Ref 228
9/16

442

IDO-17021
April 1965
Rev.

SAFETY ANALYSIS REPORT
ADVANCED TEST REACTOR
Volume 2 of 2

MASTER

D. R. deBoisblanc
S. Cohen

PHILLIPS
PETROLEUM
COMPANY

ATOMIC ENERGY DIVISION

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

NATIONAL REACTOR TESTING STATION
US ATOMIC ENERGY COMMISSION

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

Printed in the United States of America
Available from
Clearinghouse for Federal Scientific and Technical Information
National Bureau of Standards, U. S. Department of Commerce
Springfield, Virginia 22151
Price: Printed Copy \$3.00; Microfiche \$0.65

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

IDO-17021
AEC Research and Development Report
Reactor Technology
TID-4500 (39th Ed.)
Issued: April 1965

SAFETY ANALYSIS REPORT

ADVANCED TEST REACTOR

VOLUME 2 OF 2

D. R. deBoisblanc
ASSISTANT MANAGER
ATOMIC ENERGY DIVISION

S. Cohen

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

PHILLIPS
PETROLEUM
COMPANY

Atomic Energy Division

Contract AT(10-1)-205

Idaho Operations Office

U. S. ATOMIC ENERGY COMMISSION

DISTRIBUTION OF THIS DOCUMENT IS UNLIMITED

ACKNOWLEDGEMENTS

Grateful acknowledgement is made to Ebasco Services Incorporated and the Babcock and Wilcox Company for the preparation of a large part of the work included in this report.

The Thermal Hydraulic and Engineering Experiments Groups of the Reactor Engineering Branch of Phillips Petroleum Company, Atomic Energy Division, provided thermal hydraulic and stress analyses related to core performance and fuel element behavior and the Critical Facility Section provided experimental data related to Reactor Physics and Nuclear Safety.

Further acknowledgement is made to the Idaho Operations Office of the United States Atomic Energy Commission for material and constructive criticism and for major content in preparing the Hazards Section of this report.

SAFETY ANALYSIS REPORT
ADVANCED TEST REACTOR

CONTENTS

<u>ACKNOWLEDGMENTS</u>	ii
I. <u>INTRODUCTION AND SUMMARY</u>	1
I-1. INTRODUCTION	3
1. SUMMARY DESCRIPTION	3
2. IDENTIFICATION OF CONTRACTORS	3
I-2. PRINCIPAL CHARACTERISTICS	4
1. REACTOR BUILDING	4
2. REACTOR SYSTEM, SHIELDING, AND EXPERIMENTS	4
3. CORE	5
4. PRIMARY COOLANT SYSTEM	6
5. INSTRUMENTATION AND ELECTRICAL SYSTEMS	6
6. AUXILIARY SYSTEMS	6
7. DESIGN AND EXPECTED PERFORMANCE CHARACTERISTICS	6
8. RESEARCH AND DEVELOPMENT PROGRAMS	12
I-3. SAFETY PRINCIPLES	13
1. OPERATION AND ADMINISTRATIVE CONTROL	13
2. ATR SAFETY FEATURES	13
I-4. SUMMARY	16
1. NORMAL OPERATION	16
2. ABNORMAL REACTIVITY ADDITIONS	16
3. OTHER ABNORMAL INCIDENTS	16
4. ENVIRONMENTAL HAZARDS	17

I-5. CONCLUSIONS	18
II. <u>ATR SITE</u>	19
II-1. SITE DESCRIPTION	21
1. LOCATION AND POPULATION	21
1.1 NRTS Environs	21
1.2 On NRTS	21
1.3 Estimated Population Growth	23
2. METEOROLOGY	26
2.1 General	26
2.2 Precipitation	26
2.3 Temperature	26
2.4 Barometric Variations	27
2.5 Lapse and Inversion Conditions	27
2.6 Winds	27
2.7 Summary	29
3. GEOLOGY AND SEISMOLOGY	36
3.1 Geology of Site	36
3.2 Seismology	36
4. HYDROLOGY	36
5. WEATHER MONITORING	37
II-2. REFERENCES	38
III. <u>REACTOR BUILDING</u>	39
III-1. BUILDING ARRANGEMENT	41
1. GENERAL DESCRIPTION	41
1.1 Casttight Areas	41
1.2 Reactor Area	41
1.3 Storage Canal Area	62
1.4 Utility Area	62
1.5 General Storage and Laydown Area	62
1.6 Critical Facility Area	63
1.7 Counting Room	63
1.8 Offices	63
1.9 First Basement	63
1.10 Second Basement	64
2. SPECIAL FEATURES	64
2.1 Building Fire Protection Systems	64
2.2 Building Canals	65
2.3 Building Design and Construction	68

III-2. CONTAINMENT	71
1. DESIGN CRITERIA FOR GASTIGHT AREAS	71
2. DETAILED DESCRIPTION OF GASTIGHT AREAS	71
2.1 Substructure	71
2.2 Superstructure	71
2.3 Penetrations	71
2.4 Ventilating Systems	73
2.5 Spray System	81
3. ANALYSIS OF ACCIDENTS CAUSING A PRESSURE RISE IN THE GASTIGHT AREAS	81
3.1 Accidents Causing a Pressure Rise	81
3.2 Summary of Pressure Rise Analysis	86
4. ANALYSIS OF LEAKAGE FROM GASTIGHT VOLUME	86
4.1 Factors Influencing Leakage	86
4.2 Analysis of Combined Building Leak Rate	87
4.3 Effect of Pressurizing Factors on Building Frame	87
5. BUILDING LEAK RATE TEST	88
5.1 Test Preparation	88
5.2 Equipment	88
5.3 Test Procedure	89
6. CONCLUSIONS	89
III-3. REFERENCES	90
IV. <u>REACTOR STRUCTURE, SHIELDING, AND EQUIPMENT</u>	91
IV-1. REACTOR VESSEL	93
1. REACTOR VESSEL DESCRIPTION	93
1.1 General	93
1.2 Vessel Rating	93
1.3 Materials of Construction	93
1.4 Top Head, Cover Plate, and Seal Arrangement	95
1.5 Bottom Head, Cover Plate, and Seal Arrangement	95
1.6 Penetrations	96
1.7 Attachments	96
1.8 Vessel Support	97
1.9 Flushing, Venting, and Overflow	97
2. DESIGN CRITERIA	97
2.1 Applicable Codes	97
2.2 Stress Analysis	99
2.3 Cyclic Behavior	99
2.4 Radiation Damage Analysis	100
2.5 Fabrication	101
2.6 Corrosion	101

3. VESSEL INTERNALS	101
3.1 Description of Vessel Internals	101
3.2 Assembly Mock-Up	108
IV-2. SHIELDING	109
1. GENERAL DESCRIPTION	109
2. REACTOR VESSEL	109
2.1 Internal Shielding	109
2.2 Nozzle Trench	109
2.3 Top-Head Shield	110
2.4 Primary Biological Shield	110
3. PRIMARY SYSTEM SHIELDING	110
3.1 Coolant Loop Shielding	110
4. SUBPILE AND ROD ACCESS ROOMS	111
5. CANAL SHIELDING	112
6. EXPERIMENTAL LOOP CUBICLES	112
7. DEMINERALIZER VAULTS	113
8. SUMMARY OF DOSE RATES	113
8.1 Dose Rates from the Release of One ATR Fuel Element to the Primary System	113
8.2 Radiation Control Areas	113
9. HANDLING OPERATIONS	115
IV-3. EXPERIMENTAL FACILITIES	116
1. GENERAL DESCRIPTION	116
2. PRESSURIZED WATER LOOP FACILITIES	116
2.1 General Description	116
2.2 Pressure Tube Assembly	117
2.3 Out-of-Pile Facilities	120
3. GAS-COOLED LOOP FACILITY	128
3.1 General Description	128
3.2 In-Pile Tube Assembly	128
3.3 Out-of-Pile Equipment	130
3.4 Instrumentation and Reactor Control	131
4. INNER AND OUTER REFLECTOR CAPSULE FACILITIES ..	133

4.1 Inner Reflector Capsule Facilities	133
4.2 The Outer Reflector Irradiation Facilities	133
IV-4. REFERENCES	134
V. <u>REACTOR CORE</u>	135
V-1. CORE DESCRIPTION	137
1. GENERAL CORE CONFIGURATION AND SUPPORT	137
1.1 Description of Components and Method of Support	137
1.2 Tolerance Analysis	139
2. FUEL ELEMENTS	141
2.1 Materials	143
2.2 Description of ATR Fuel Element Fabrication Techniques	143
2.3 Fuel Element Hydraulic Development	145
2.4 ATR Fuel Element Stress Analysis	150
2.5 Irradiation Test Results	157
2.6 Corrosion Tests	158
2.7 Chemical Reactions Between Aluminum and U_3O_8 Dispersions	159
2.8 Summary	161
3. REFLECTOR AND MODERATOR	162
3.1 Description	162
3.2 Cooling of Reflector	162
3.3 Stress Analysis and Clearances	162
3.4 Corrosion and Radiation Effects Studies	162
4. CONTROL AND SAFETY ELEMENTS	164
4.1 General Description	164
4.2 Safety Rods	164
4.3 Neck Shim Rods	165
4.4 Outer Shim Drums	165
4.5 Hafnium Performance	165
V-2. NUCLEAR CHARACTERISTICS	167
1. REACTOR PHYSICS	167
1.1 Reactivity Distribution and Control Worth	167
1.2 Movable Control	169
1.3 Safety Rods	170
1.4 Temperature and Void Coefficients	170
1.5 Neutron Flux Distribution	174
1.6 Neutron Lifetime and Delayed Neutron Fraction	178
2. KINETIC NUCLEAR CHARACTERISTICS	180
2.1 General	180
2.2 Safety Rods	180

2.3 Neck Shim Control Rods	179
2.4 Outer Shim Control Cylinders	180
2.5 Power Regulation	181
2.6 Safety Action Response	181
2.7 Xenon Stability	184
2.8 Interlobe Coupling	184
V-3. THERMAL AND HYDRAULIC CHARACTERISTICS	193
1. GENERAL	193
2. CORE HYDRAULIC CHARACTERISTICS	193
3. THERMAL-HYDRAULIC ANALYSIS	195
3.1 Hot Spot-Hot Channel Analysis	195
3.2 Statistical Analysis	196
3.3 Thermal Burnout Analyses	202
3.4 Low Flow and Decay Heat Removal	203
V-4. REFERENCES	205
VI. <u>PRIMARY COOLANT SYSTEM</u>	207
VI-1. SYSTEM ARRANGEMENT AND CONTROL	209
1. GENERAL	209
2. METHODS OF CONTROL	209
2.1 Flow Control	209
2.2 Temperature Control	209
2.3 Pressure Control	209
2.4 Coolant Quality Control	211
2.5 Emergency Cooling	211
VI-2. DESIGN OF PRINCIPAL COMPONENTS	213
1. GENERAL	213
2. PRIMARY HEAT EXCHANGERS	213
2.1 Arrangement	213
2.2 Design Characteristics	213
2.3 Materials of Construction	214
2.4 Codes and Tests	214
3. PRIMARY COOLANT PUMPS	214
3.1 Arrangement	214
3.2 Design Characteristics	215
3.3 Materials of Construction	215
3.4 Codes and Tests	215
4. EMERGENCY COOLANT PUMPS	216
4.1 Arrangement	216

4.2 Design Characteristics	217
4.3 Materials of Construction	218
4.4 Codes and Tests	218
5. PRIMARY PRESSURIZING PUMPS	219
5.1 Arrangement	219
5.2 Design Characteristics	219
5.3 Materials of Construction	220
5.4 Codes and Tests	220
6. SURGE TANK	220
6.1 Design Characteristics	220
6.2 Materials of Construction	221
6.3 Codes and Tests	221
7. DEGASSING TANK	221
7.1 Design Characteristics	221
7.2 Materials of Construction	221
7.3 Codes and Tests	221
8. PIPING	222
8.1 General Arrangement -- Materials of Construction -- Codes	222
8.2 Welding	222
8.3 Primary Coolant System Pipe Stresses	222
8.4 Flexibility and Methods of Support	223
8.5 Vibration Analysis	223
9. VALVES	223
VI-3. PRIMARY COOLANT RELATED SYSTEMS	226
1. SURGE TANK AND PRESSURIZING SYSTEM	226
2. PRESSURE RELIEF SYSTEM	226
3. PRIMARY COOLANT pH CONTROL SYSTEM	226
4. BYPASS DEMINERALIZER SYSTEM	227
4.1 System Description	227
4.2 System Capacity	227
4.3 Materials of Construction	227
4.4 Codes and Tests	229
4.5 Operating Limitations	229
5. DEGASSING SYSTEM	229
5.1 System Description	229
5.2 Gas Removal Design and Capacity	229
5.3 Activity Monitoring	230

6. GLAND SEALING SYSTEM	231
6.1 System Description	231
6.2 System Supplied with Seal Water	231
6.3 System Leak Detection	231
7. DEMINERALIZED WATER MAKEUP SYSTEM	233
7.1 System Description	233
7.2 System Uses	233
8. REACTOR FLUSH SYSTEM	233
9. EMERGENCY WATER SUPPLY	235
9.1 Sources	235
9.2 Sequence of Events Following Large Leakage	235
9.3 Description of Fire-Water System	236
9.4 Analysis of Large Ruptures in the Primary System Handled By Emergency Water Supply	236
VII. <u>INSTRUMENTATION AND CONTROL</u>	239
VII-1. CONTROL METHODS AND PHILOSOPHY	241
VII-2. NUCLEAR INSTRUMENTATION AND CONTROL	242
1. GENERAL	242
2. NEUTRON DETECTION	243
2.1 Start-Up Instrumentation	243
2.2 Log-N Period Channels	245
2.3 Neutron Level Channels	245
2.4 Sigma Bus System	245
2.5 Deviation Galvanometer	246
2.6 Servo System	246
2.7 Neutron Detector Locations	246
2.8 Nuclear Instrumentation Nonlinearities	246
3. N-16 MONITORING SYSTEM	248
4. FISSION BREAK MONITORS	250
VII-3. REACTOR CONTROL	251
1. CONTROL PHILOSOPHY	251
2. METHODS OF CONTROL	251
2.1 Outer Shim Control Cylinders	251
2.2 Neck Shim Control Rods	251
2.3 Servo or Automatic Control	251
2.4 Automatic Power Reduction	254

3. CONTROL INTERLOCKS	254
3.1 Operational Sequence Interlocks	254
3.2 Console Operation Interlocks	256
3.3 Safety Rod Interlocks	256
3.4 Outer Shim Cylinder Interlocks	256
3.5 Neck Shim Rod Interlocks	257
3.6 Regulating Rod Interlocks	257
3.7 Control Panel Instrument Interlocks	257
4. LOW-POWER, OPEN-HEAD OPERATION	261
VII-4. REACTOR AND PLANT SAFETY SYSTEM	262
1. GENERAL	262
2. TYPES OF ACCIDENTS	262
2.1 Malfunction of Reactor Equipment	262
2.2 Plant Equipment Malfunction	262
2.3 Experiment Failures	263
3. SAFETY SYSTEM ACTION	264
4. SAFETY SYSTEM INPUTS	264
5. SHIM INSERT DELAY TIMER	272
5.1 Operation of Timer	272
5.2 Possible Accidents	273
VII-5. PROCESS INSTRUMENTATION AND CONTROL	274
1. GENERAL	274
2. PRIMARY COOLANT SYSTEM INSTRUMENTATION	274
2.1 General	274
2.2 Primary Coolant Temperature Control System	278
2.3 Pressure Control and Pressurizing System	280
2.4 Water Power Measurements	281
2.5 Primary Emergency Coolant Pump Instrumentation	284
2.6 Pump Bearing Temperatures	284
2.7 Gland Seal and Bottom-Head Flush System	284
2.8 Primary Degassing Tank	286
2.9 Bypass Demineralizer Instrumentation -- Normal Operation	287
2.10 pH Control Primary and Secondary Coolant Systems	288
2.11 Reactor Shielding Temperatures	288
2.12 Reactor Vessel Liquid Level Instrumentation	288
VII-6. RADIATION MONITORING SYSTEM (RMS)	290
1. GENERAL	290
2. BETA-GAMMA MONITORS	291

3. AIRBORNE PARTICULATE MONITORING SYSTEM	293
4. HAND AND FOOT MONITORS	294
5. PORTAL MONITORS.	295
6. ANALYTICAL GAS MONITOR	296
7. IODINE TYPE RUPTURED FUEL ELEMENT MONITOR . . .	296
8. INTERLOCK RELAYS FOR H AND V SYSTEM	297
9. EXPERIMENTAL CUBICLE MONITORING	298
9.1 Area	298
9.2 Process	298
VII-7. DATA PROCESSING SYSTEM.	300
1. NORMAL SCANNING.	300
1.1 Comparison of Analog Inputs Against Limits	300
1.2 Routine Periodic Logging	300
1.3 Rapid Computation of Predetermined Quantities	300
1.4 Comparison of Computed Quantities Against Limits . . .	300
2. INDEPENDENT LOGIC CHECK OF THE OPERATIONS OF THE CONTROL SYSTEM RELAY NETWORK	300
3. MEMORY	301
4. ACCELERATED SCANNING DURING TRANSIENTS	301
5. CALCULATION AND INTERPRETATION OF INFORMATION	301
VII-8. ATR SYSTEM TRANSIENT BEHAVIOR	303
1. CLOSED LOOP ANALOG STUDY OF THE ATR	303
2. SERVO REGULATOR ROD TEST PROGRAM	305
VIII. <u>CONTROL ROD DRIVES</u>	307
VIII-1. SAFETY ROD DRIVES	309
1. GENERAL DESCRIPTION	309
1.1 Location and Arrangement	312
1.2 Operating Characteristics	312
1.3 Materials of Construction	312
1.4 Access	313
1.5 Seals	313
1.6 Cooling	313
1.7 Drive System	314

1.8 Position Indication	314
1.9 Scram Action	314
2. DESIGN CONSIDERATIONS	315
2.1 Rack and Pinion	315
2.2 Outer Snubber Tube	316
2.3 Rack Tube Guide Bushings	316
2.4 Horizontal Shaft Ball Bearings	316
2.5 Scram Spring	316
2.6 Ball Nut and Screw Assembly	316
2.7 Electric Clutch	316
2.8 Rotary Seal	317
2.9 Misalignment of Components	317
3. COMPONENT MALFUNCTION	317
4. PROTOTYPE SAFETY ROD DRIVE ASSEMBLY TESTS	318
4.1 Operational Testing	318
4.2 Life Tests	319
4.3 Miscellaneous Tests	319
5. SUMMARY	322
VIII-2. NECK SHIM AND REGULATING ROD DRIVE MECHANISMS . . .	323
1. GENERAL DESCRIPTION	323
1.1 General Characteristics of the Neck Rods	326
1.2 Materials of Construction	326
1.3 Access and Maintenance	328
1.4 Cooling	328
1.5 Position Indication	328
1.6 Drive Power Unit	329
1.7 Seals	329
2. DESIGN CONSIDERATIONS	330
2.1 Flexible Cable Drive Section	330
2.2 Regulating Rod Drive	330
2.3 Pressure Containing Members	330
3. COMPONENT MALFUNCTION	331
3.1 Neck Shim Drive Mechanism	331
3.2 Neck Regulating Drive Mechanism	331
3.3 Neck Regulating Shim Drive Mechanism	332
4. NECK SHIM AND REGULATING ROD DRIVE PROTOTYPE TESTS	332
4.1 Neck Shim Rod Cable Drive Mechanism	333
4.2 Neck Regulating Rod Mechanism	333
5. SUMMARY	334

VIII-3. OUTER SHIM CONTROL DRIVE MECHANISM	335
1. GENERAL DESCRIPTION	335
1.1 Operating Characteristics	337
1.2 Materials of Construction	337
1.3 Access and Maintenance	337
1.4 Seal	338
1.5 Cooling	338
1.6 Position Indication and Limit Control.	338
2. DESIGN CONSIDERATIONS	338
3. MALFUNCTIONS	339
3.1 Electrical Power Failure.	339
3.2 Limit Switch Malfunction	339
3.3 Failure of External or Internal Mechanical Stop in Conjunction with Limit Switch Failure	339
3.4 Excessive Leakage of Shaft Seal	340
3.5 Seizure of Shaft Seal	340
3.6 Seizure of Drive-Shaft Bearings	340
3.7 Seizure of the Worm Gears or Bearings	340
3.8 Failure of the Coupling Bolt	340
3.9 Breakage of the Coupling-Shaft Spring	340
4. PROTOTYPE OUTER SHIM CONTROL ROD DRIVE TESTS	340
4.1 Life Cycle Test	341
4.2 Environment Tests	341
4.3 Misalignment Tests	341
4.4 Position Accuracy Tests	341
5. SUMMARY	342
IX. <u>ELECTRICAL SYSTEMS</u>	343
IX-1. POWER SYSTEM ARRANGEMENT	345
1. ELECTRICAL DISTRIBUTION FACILITIES	345
1.1 General Description	345
1.2 Commercial Power	346
1.3 Diesel Power.	347
1.4 Diesel-Commercial Power.	348
1.5 Instrumentation AC Battery-Backed Power	348
1.6 Utility DC Power	348
1.7 Existing Power System Extensions	349
1.8 Emergency Lighting	349
IX-2. POWER SYSTEM PROTECTION	355
1. LIGHTNING PROTECTION.	355
2. OVERLOAD PROTECTION	355
2.1 ETR Transformers	355
2.2 Conductors	355
2.3 7500-kVA ATR Transformers	356

2.4 Switching Equipment	356
2.5 Motors	356
2.6 Diesel Generators	356
3. SERVICE RELIABILITY	356
4. RESTORATION OF ESSENTIAL SERVICES AFTER BUILDING EVACUATION	358
4.1 General Description	358
4.2 Equipment Automatically Restored and Manually Started	358
IX-3. ALARMS AND COMMUNICATION	361
1. ALARM SYSTEMS	361
1.1 Fire Alarm	361
1.2 Evacuation Alarms	361
1.3 Reactor Alarms	361
1.4 Other Alarms	361
2. COMMUNICATION SYSTEMS	361
2.1 Telephone	361
2.2 Paging	361
2.3 DC Powered Telephone	361
X. <u>ADDITIONAL COOLANT SYSTEMS</u>	363
X-1. SECONDARY COOLANT SYSTEM	365
1. GENERAL	365
2. WATER TREATMENT	367
3. DETECTION OF PRIMARY HEAT-EXCHANGER LEAKS	367
4. COOLING TOWER MAKEUP	367
5. EQUIPMENT DESIGN	367
5.1 General	367
5.2 Equipment Performance	367
X-2. UTILITY COOLING-WATER SYSTEM (UCW)	370
1. PURPOSE AND ARRANGEMENT	370
2. DESIGN FEATURES	370
3. SPECIAL FEATURES	370
X-3. HIGH PRESSURE DEMINERALIZED WATER SYSTEM (HDW)	372
1. PURPOSE AND ARRANGEMENT	372

2. DESIGN FEATURES	372
X-4. SECONDARY COOLANT SYSTEM INSTRUMENTATION	374
1. GENERAL	374
2. COOLING TOWER MAKEUP CONTROL	374
3. SECONDARY COOLANT BLOWDOWN CONTROL	374
4. PRIMARY HEAT EXCHANGER SECONDARY SIDE TEMPERATURE	374
5. SECONDARY COOLANT-WATER POWER MEASUREMENT	375
6. UTILITY COOLING-WATER SYSTEM INSTRUMENTATION	375
7. HIGH PRESSURE DEMINERALIZED WATER (HDW) SYSTEM INSTRUMENTATION.	376
7.1 HDW Expansion Tank	376
7.2 HDW Circulating Pumps	376
7.3 HDW Heat Exchangers	377
XI. <u>AUXILIARY SYSTEMS AND MACHINERY</u>	379
XI-1. REACTOR AUXILIARY SYSTEMS	381
1. FUEL ELEMENT HANDLING	381
2. CONTROL COMPONENT HANDLING	381
2.1 Safety Rod Assembly	381
2.2 Neck Shim and Regulating Rod Drive	383
2.3 Outer Shim Control Handling	383
3. INTERNALS HANDLING	384
3.1 Vibration Ties	384
3.2 Neck Shim Housing Pressure Tube Locator Ties	384
3.3 Beryllium Reflector	384
3.4 Aluminum Reflector	384
3.5 Neck Shim Rod Housing	384
3.6 Neck and Regulating Rod Sleeve	384
3.7 The Flux Trap Baffle	384
3.8 Drop Tube Cover	385
3.9 Reactor Internals Protective Covers	385
4. EXPERIMENT HANDLING	385
4.1 Loop Handling	385
4.2 Capsules	387
XI-2. FACILITY AUXILIARY SYSTEMS	388
1. PLANT AND INSTRUMENT AIR SYSTEMS	388
2. HIGH PRESSURE AIR SYSTEM	388

3. DIESEL GENERATOR AUXILIARY SYSTEMS	388
3.1 Diesel Oil Storage	388
3.2 Cooling System	388
3.3 Lube Oil System	388
3.4 Emergency Starting	389
3.5 System Instrumentation	389
4. PLANT HEATING AND PROCESS STEAM SYSTEM	390
5. POTABLE WATER SYSTEM	390
XI-3. WASTE DISPOSAL	391
1. HOT WASTE	391
1.1 General Description	391
1.2 Instrumentation and Sampling	391
1.3 Materials of Construction	391
2. WARM WASTE	393
2.1 General Description	393
2.2 Instrumentation and Sampling	393
2.3 Materials of Construction	393
3. COLD WASTE	394
4. SANITARY WASTE	394
5. SOLID WASTE	394
6. GASEOUS WASTE	394
XI-4. FIRE PROTECTION SYSTEM	395
XII. <u>COUPLING BETWEEN REACTORS IN THE MTR-ETR-ATR</u> <u>AREA</u>	397
XII-1. GENERAL CONSIDERATIONS	399
XII-2. COUPLING BY DIRECT RADIATION DURING NORMAL OPERATION	400
XII-3. METEOROLOGICAL COUPLING DURING NORMAL OPERATION .	401
XII-4. COUPLING THROUGH COMMON FACILITIES	402
1. COMMON SYSTEMS INDIRECTLY AFFECTING REACTOR OPERATIONS	402
1.1 Chemical Storage	402
1.2 Diesel Oil Storage	403
1.3 Potable Water System	403
1.4 Heating Steam and Condensate	403
1.5 Liquid Waste Disposal	403

1.6 Communications	403
2. SYSTEMS DIRECTLY AFFECTING REACTOR OPERATION	404
2.1 Commerical Electric Power System	404
2.2 Instrument and Plant Air System	405
2.3 Raw Water System	405
2.4 Fire-Water System	406
2.5 Demineralized Water System	406
2.6 Evacuation and Fire Alarms	407
XII-5. SUMMARY	408
XII-6. REFERENCE	409
XIII. <u>EMERGENCY EVACUATION PROCEDURES</u>	411
XIII-1. EVACUATION	413
1. GENERAL	413
2. EVACUATION OF THE MTR-ETR-ATR SITE	413
3. EVACUATION OF OTHER NRTS SITES	413
4. EVACUATION OF OFF-SITE AREAS	415
XIII-2. ATR EVACUATION PLAN	416
1. ETR EVACUATION PLAN	416
2. TRA EMERGENCY ACTION PLAN	421
XIII-3. REFERENCE	426
XIV. <u>INITIAL TESTS AND OPERATING PROCEDURES</u>	427
XIV-1. PREOPERATIONAL SYSTEMS TESTS	429
1. COMPONENT CONTINUITY TESTS	429
2. SYSTEMS OPERATION TEST	430
XIV-2. NEUTRON TESTS	432
1. PRELOADING PROCEDURES AND TESTS	432
2. CRITICALITY TESTS AND FULL CORE LOADING FOR THE ATR	432
3. POST NEUTRON, LOW POWER PHYSICS TESTS FOR THE ATR	433
4. APPROACH TO POWER	434

5. ATTAINMENT OF FULL POWER -- CORE I	435
6. CORE II TESTS	435
XIV-3. OPERATING PROCEDURES (PRELIMINARY)	436
XV. <u>ADMINISTRATIVE AND PROCEDURAL SAFEGUARDS</u>	447
XV-1. ORGANIZATION AND FUNCTION	449
1. PHILLIPS PETROLEUM COMPANY AED ORGANIZATION STRUCTURE	449
2. ATR OPERATIONS STRUCTURE	449
3. SUPPORT ORGANIZATIONS	452
3.1 Reactor Engineering Branch	452
3.2 Mathematical Analysis and Machine Computations Branch	452
3.3 Instrument Development and Technical Services Branch	453
3.4 Division Engineering	453
4. ATR START-UP COMMITTEE	453
5. REACTOR SAFEGUARD COMMITTEE	454
6. SAFEGUARD REVIEW COMMITTEE	455
7. NUCLEAR SAFETY COMMITTEE	455
8. HEALTH AND SAFETY	455
9. ACCIDENT REVIEW COMMITTEE	456
10. PROJECT ENGINEERING	456
11. TEST REACTOR OPERATIONS TRAINING AND QUALIFICATIONS COMMITTEE	457
XV-2. ADMINISTRATIVE CONTROLS AND PROCEDURES	458
1. STANDARD PRACTICES MANUAL	458
2. OPERATING MANUALS	458
3. MAINTENANCE CONTROL	458
XV-3. REVIEW AND AUTHORIZATION OF TESTS, EXPERIMENTS, MODIFICATIONS, AND SAFETY STATUS	460
1. PROCEDURES AND AUTHORIZATION FOR PERFORMING REACTOR TESTS	460

2. PROCEDURES AND AUTHORIZATION FOR INSERTING EXPERIMENTS	460
3. PROCEDURES AND AUTHORIZATION FOR MODIFYING FACILITY	460
4. REVIEW AND SURVEILLANCE OF REACTOR SAFETY QUESTIONS	461
5. ATOMIC ENERGY COMMISSION AUTHORIZATION	461
XV-4. TRAINING	462
1. OPERATIONAL TRAINING	462
1.1 Formal Training	462
1.2 Practical Training	462
1.3 Qualifications	463
2. FIRE-SAFETY-RADIATION TRAINING	463
XV-5. SUMMARY	464
XVI. <u>ACCIDENT ANALYSIS</u>	465
XVI-1. REACTIVITY ADDITIONS	467
1. GENERAL	467
2. ACCIDENTAL REACTIVITY ADDITION FROM CONTROL ELEMENTS	467
2.1 Analysis of Possible Causes	467
2.2 Start-Up Accident	468
2.3 Rod Withdrawal Accident at Power	470
2.4 Experiment Feedback	472
3. OTHER SOURCES OF REACTIVITY ADDITION	473
3.1 Fuel Element Collapse	473
3.2 Damage to the Beryllium Reflector	474
3.3 Loss of Specimen in the In-Pile Loop	474
3.4 Loss of Reflector Water	474
3.5 Fuel Handling Accident	474
3.6 Cold Water Accident	474
3.7 Reactivity from the Experiment	475
4. EFFECT OF MAXIMUM REACTIVITY ADDITION	475
4.1 Model	476
4.2 Reactor at Full Power	477
4.3 Open-Head Operation	477
5. SUMMARY OF REACTIVITY ADDITIONS	478
XVI-2. NONNUCLEAR ACCIDENTS	479

1. LOSS-OF-COOLANT-FLOW ACCIDENT.	479
1.1 General	479
1.2 Safety Action	479
1.3 Description of Calculations	480
1.4 Results	480
1.5 Summary	481
2. LOSS-OF-COOLANT ACCIDENT.	484
2.1 General	484
2.2 External Leaks	484
2.3 Internal Leak.	486
3. LOSS OF POWER	487
3.1 Effect of Electrical Power Failures on Operation	487
3.2 Effect of Loss of Power on Shim Controls	489
4. FUEL ELEMENT FAILURES	493
4.1 Fuel Element Failures in the Primary System	493
4.2 Flow Blockage Accident	494
4.3 Cladding Failures During Fuel Element Handling Operations	495
5. REACTOR MISSILE PRODUCTION	495
5.1 Reactor Top Shield as a Missile	495
5.2 Transfer Shield Plate, Shield Plug, or Loop Closure Plug as a Missile	497
5.3 Production of Missiles from Reactor Internals	498
6. FUEL TRANSFER ACCIDENT	499
7. PRESSURIZING SYSTEM FAILURES	499
7.1 Air Inventory Accidents in Surge Tank	500
7.2 Failure of Surge Tank Nozzles and Penetrations	500
7.3 Failure of Pressurizing Pump System	500
7.4 Conclusions	502
8. EXPERIMENT FAILURES	502
8.1 General	502
8.2 Safety Evaluation of the Loop Closure Plug	502
8.3 Evaluation of Loop In-Pile Pressure Tube	503
8.4 Gas Loop Failure	503
9. MISCELLANEOUS ACCIDENTS	504
9.1 Relief Valve Stuck Open	504
9.2 Earthquake	504
XVI-3. REACTOR DAMAGE EVALUATION	505
1. GENERAL	505

2. FUEL MELTING DURING A NUCLEAR EXCURSION	506
3. FUEL MELTING RESULTING FROM A LARGE PRIMARY SYSTEM RUPTURE	509
4. FUEL MELTING DURING A FLOW BLOCKAGE ACCIDENT	513
5. REACTOR VESSEL INTEGRITY	513
5.1 General	513
5.2 Slow Pressure Rise	514
5.3 Explosive Pressure Rise	517
5.4 Summary	519
XVI-4. REFERENCES	520
XVII. <u>HAZARDS ANALYSIS</u>	521
XVII-1. MINOR RADIOLOGICAL HAZARDS	523
1. GENERAL	523
2. FISSION BREAKS INTO THE PRIMARY COOLANT	523
3. FISSION BREAKS INTO EXPERIMENT LOOP COOLANT	523
4. RELEASE OF RADIOACTIVE GAS INTO THE BUILDING AIR	524
5. RELEASE OF SOLID OR LIQUID RADIOACTIVE MATERIALS INTO THE BUILDING	524
XVII-2. ENVIRONMENTAL HAZARDS ASSOCIATED WITH NORMAL OPERATION	525
XVII-3. ENVIRONMENTAL HAZARDS ASSOCIATED WITH LARGE RELEASES OF RADIOACTIVITY	526
1. GENERAL	526
2. FUEL TRANSFER ACCIDENT	526
2.1 General	526
3. MAXIMUM HYPOTHETICAL ACCIDENT	527
3.1 General	527
3.2 Time Scale of Accident	528
3.3 Conservatism in Analysis	528
XVII-4. DOSE CALCULATIONS	531
1. GENERAL	531

2. METEOROLOGICAL FACTORS	531
2.1 Short Term Adverse Weather	532
2.2 Long Term Adverse Weather	532
3. CONSERVATISM IN ANALYSIS	535
4. DOSE RATES AT MTR-ETR-ATR AREA	536
4.1 General	
4.2 Fuel Handling Accident	536
4.3 Maximum Hypothetical Accident	540
5. DOSE RATES UP TO 15 MILES OUTSIDE MTR-ETR-ATR AREA	547
5.1 General	547
5.2 Fuel Handling Accident	548
5.3 Maximum Hypothetical Accident	549
6. INTEGRATED DOSES UP TO 50 MILES FROM MTR-ETR- ATR AREA	550
XVII-5. THE CONSEQUENCES OF AN ATR ACCIDENT	553
1. CONSEQUENCES OF AN ATR ACCIDENT TO THE MTR- ETR-ATR AREA	553
2. HAZARDS WITHIN NRTS BEYOND THE MTR-ETR-ATR AREA	554
3. AREA ACCESSIBILITY AFTER ACCIDENT BEYOND MTR-ETR AREA	555
4. HAZARDS TO AREAS OUTSIDE NRTS	556
4.1 Low Population Area	556
5. CONCLUSION	556
XVII-6. ENVIRONMENTAL HAZARDS ASSOCIATED WITH LARGE RELEASES OF RADIOACTIVITY IN THE NON-GASTIGHT AREA	558
XVII-7. REFERENCES	560
APPENDIX A -- <u>IRRADIATION EFFECTS ON BERYLLIUM</u>	561
APPENDIX B -- <u>ENVIRONMENTAL HAZARDS</u>	569
1. NOMENCLATURE FOR ENVIRONMENTAL HAZARDS ANALYSIS	571
2. GENERAL DATA	573
3. INTEGRATED IODINE DOSE TO THE THYROID	574

4. IODINE DOSE RATE TO THE THYROID	574
5. INTEGRATED BETA DOSE FROM THE PLUME	575
6. BETA DOSE RATE FROM THE PLUME	576
7. INTEGRATED GAMMA DOSE FROM THE PLUME	577
8. GAMMA DOSE RATE FROM PLUME	578
9. DIRECT GAMMA DOSE RATE	579
10. REFERENCE	579
APPENDIX C -- <u>ATR FISSION PRODUCT INVENTORY AFTER 17 DAYS</u> <u>OF OPERATION</u>	581
APPENDIX D -- <u>REACTOR BUILDING LEAK RATE COMPUTATIONS</u> . .	589
1. CONSTANTS	591
2. ASSUMPTIONS	591
3. FORMULAS	591
4. COMPUTATIONS	592
5. REFERENCES	596
APPENDIX E -- <u>RESEARCH AND DEVELOPMENT PROGRAMS</u>	597
1. BURNOUT HEAT TRANSFER TESTS	599
2. HYDRAULIC TESTS	599
3. PROCEDURES FOR FABRICATING ALUMINUM-BASE ATR FUEL ELEMENTS	600
4. FUEL IRRADIATION TESTS	600
5. FUEL ELEMENT CLADDING CORROSION TESTS	600
6. MATERIALS OF CONSTRUCTION TESTS	600
7. THERMITE REACTION EVALUATION PROGRAM	600
8. ATR CRITICAL EXPERIMENT FINAL REPORT	601
9. MECHANICAL MOCK-UP TESTS	601
10. SERVO REGULATOR ROD TESTS	601
11. DESIGN STUDIES	601
APPENDIX F -- <u>ERROR PROPAGATION ANALYSIS OF THERMAL</u> <u>BURNOUT</u>	603

FIGURES

II-1-1. Map of National Reactor Test Station and adjacent areas	22
II-1-2. Plan of National Reactor Test Station	24
II-1-3. ATR-MTR-ETR plot plan	25
II-1-4. WBO average thermal gradients between 5- and 250-foot levels and associated N-values, January 1955 to May 1958 ...	28
II-1-5. WBO seasonal variations in the average thermal gradient between 5-foot and 250-foot levels, September 1950 through August 1952	28
II-1-6. Time of fumigation WBO -- July 1951 through December 1953	29
II-1-7. CF 250-foot level, wind roses -- July 1961 through May 1962 ..	30
II-1-8. CF 250-foot level, lapse wind roses -- July 1961 through May 1962	31
II-1-9. CF 250-foot level, inversion wind roses -- July 1961 through May 1962	32
II-1-10. CF 20-foot level, wind roses -- January 1950 through May 1962	33
II-1-11. CF 20-foot level, lapse wind roses -- January 1950 through May 1962	34
II-1-12. CF 20-foot level, inversion wind roses -- January 1950 through May 1962	35
III-1-1. Reactor building elevations	42
III-1-2. Reactor building elevations	43
III-1-3. Reactor building section	44
III-1-4. Reactor building section	45
III-1-5. Reactor building sections	46
III-1-6. First floor plan	47
III-1-7. First and second mezzanine floors	48
III-1-8. First basement floor plan	49
III-1-9. Second basement floor plan	50

III-1-10. Gastight area -- first floor plan	51
III-1-11. Gastight area -- mezzanine plan	52
III-1-12. Gastight area section	53
III-1-13. Gastight area section	54
III-1-14. Gastight area section	55
III-1-15. Gastight area section	56
III-1-16. Gastight area -- first basement plan	57
III-1-17. Gastight area -- second basement plan	58
III-1-18. Gastight area -- miscellaneous plans	59
III-1-19. Gastight area -- penetration details	60
III-1-20. Gastight area wall sections	61
III-1-21. Reactor building live-load diagrams	69
III-2-1. H and V schematic diagram	74
III-2-2. Block diagram volumes of various gastight areas	82
IV-1-1. ATR vessel elevation	94
IV-1-2. ATR vessel cutaway	95
IV-1-3. Stress analysis locations	99
IV-1-4. Vessel internals elevation	102
IV-1-5. Vessel internal cutaway	103
IV-1-6. Vessel internals plan	104
IV-1-7. Core elevation	105
IV-3-1. In-pile loop arrangement	117
IV-3-2. Pressurized water in-pile tube arrangement	118
IV-3-3. ATR shielded cubicle and enclosed sample area	122
IV-3-4. Loop facilities flow diagram	124
IV-3-5. Loop handling arrangement	127
IV-3-6. Gas loop in-pile tube arrangement	129

IV-3-7. Gas loop flow sheet	132
V-1-1. Core arrangement plan	137
V-1-2. Schematic cross section at core midplane	138
V-1-3. Fuel element details	142
V-1-4. Fuel assembly and details	143
V-1-5. Hydraulic pressure-deflection test for plate number 19 with concave pressure at room temperature	147
V-1-6. Properties of 6061 aluminum alloy	152
V-1-7. Properties of 6061 aluminum alloy	152
V-1-8. Properties of 6061 aluminum alloy	152
V-1-9. Fuel plate thermal stress integration regions	156
V-1-10. Outer shim cylinder arrangement	163
V-2-1. Integral safety rod worth	172
V-2-2. Reactivity effect of voiding an A-1 flux trap and experiment flow tube	174
V-2-3. Power generated per element at time zero	175
V-2-4. Power profiles for plates in fuel element with hot spot	178
V-2-5. Core peak power during fuel cycle	178
V-2-6. Plate 19 midplane power profile in element containing the core hot spot	178
V-2-7. ATR response to a 0.6 percent δk -step addition at beginning of core life	182
V-2-8. Reactivity feedback to a 0.6 percent δk -step addition	183
V-2-9. ATR scram curves	185
V-2-10. Instability of extended length ATR core to axial xenon oscillations	187
V-2-11. Stability of reference core length to axial xenon oscillations	188
V-2-12. Amplitude response -- chamber position 7 -- fundamental	189
V-2-13. Amplitude response -- chamber position 3 -- fundamental	190

V-2-14. Phase response -- chamber position 3 -- fundamental	191
V-2-15. Phase response -- chamber position 7 -- fundamental	192
V-3-1. Axial temperature profiles along the hot fuel plate and coolant channel	198
V-3-2. Axial temperature profiles along the hot fuel plate and coolant channel	198
V-3-3. Axial temperature profiles along the hot fuel plate and coolant channel	199
V-3-4. Axial temperature profiles along the hot fuel plate and coolant channel	199
VI-1-1. Primary coolant system flow sheet	210
VI-2-1. Primary coolant system pump head curves	216
VI-2-2. Primary coolant system hydrodynamics	217
VI-2-3. Emergency coolant pump (670 M-10) performance curve . . .	219
VI-2-4. Emergency coolant pump (670 M-11) performance curve . . .	219
VI-2-5. Pressurizing pump (670 M-14) performance curve	220
VI-2-6. Pressurizing pump (670 M-15) performance curve	220
VI-2-7. Primary system pipe stress analysis	224
VI-2-8. Primary system pipe stress analysis	225
VI-3-1. Bypass demineralizer flow sheet	228
VI-3-2. Gland seal system flow sheet	232
VI-3-3. Demineralized water makeup system	234
VII-2-1. Power level instrument relationships	243
VII-2-2. Nuclear instrumentation	244
VII-2-3. Power level deviation monitor	246
VII-2-4. Nuclear instrumentation thimble positions	247
VII-2-5. Approach to criticality	248
VII-2-6. Monitoring instrumentation	249
VII-3-1. Regulator rod control ranges	252

VII-3-2. Power level servo system	253
VII-4-1. Safety system schematic	271
VII-4-2. ATR shim insert delay timer	272
VII-5-1. Reactor control panel arrangement	275
VII-5-2. Reactor console arrangement	276
VII-5-3. Process water panel arrangement	277
VII-5-4. Primary coolant temperature control system	279
VII-5-5. Reactor power heat balance system	282
VII-5-6. Reactor power block diagram	283
VII-8-1. ATR plant functional diagram	304
VIII-1-1. Control rod drive arrangement	310
VIII-1-2. Safety rod drive arrangement	311
VIII-1-3. ATR safety rod scram performance curve	315
VIII-1-4. Deposits in outer snubber tube after crud test	321
VIII-1-5. Deposits on front pinion bearing after crud test	322
VIII-2-1. ATR neck shim rod drives	324
VIII-2-2. Neck shim flexible cable performance curves	333
VIII-3-1. Outer shim control drive	336
IX-1-1. Main one-line wiring diagram	350
IX-1-2. Auxiliary one-line wiring diagram -- sheet #1	351
IX-1-3. Auxiliary one-line wiring diagram -- sheet #2	352
IX-1-4. 250-V DC and 120/208-V AC one-line wiring diagram	353
IX-1-5. Auxiliary one-line wiring diagram -- connections to existing facilities	354
X-1-1. Secondary coolant system flow sheet	366
X-2-1. Utility cooling-water system flow sheet	371
X-3-1. High pressure demineralized water system flow sheet	373
XI-1-1. Fuel element handling arrangement	382

XI-1-2. Loop test assembly handling arrangement	386
XI-3-1. Waste handling and disposal system	392
XV-1-1. Phillips Petroleum Company AED organization structure . .	450
XV-1-2. ATR operations organization	451
XVI-1-1. Start-up accident at 10^{-5} full power with the maximum rod withdrawal rate, 0.1 percent $\Delta\rho$ per second	469
XVI-1-2. Peak power for open-head start-up accidents at the maximum rod withdrawal rate, 0.1 percent $\Delta\rho$ per second . .	471
XVI-1-3. Rod withdrawal accident in the power range	472
XVI-1-4. Open-head rod withdrawal accident peak power	473
XVI-1-5. Maximum reactivity accident for the ATR at normal power	477
XVI-1-6. Maximum reactivity accident for the ATR open-head operation	478
XVI-2-1. System flow diagram	481
XVI-2-2. Evaluation of the loss of three primary coolant pumps	482
XVI-2-3. Evaluation of the loss of one primary coolant pump	483
XVI-2-4. Reactor flow versus time after rupture (three-inch-diameter leak in reactor inlet line, 355 psig)	486
XVI-2-5. Electrical power sources and loads	488
XVI-2-6. Xenon transient (100 percent to zero power)	490
XVI-2-7. Shim and safety rod drive power supply	492
XVI-2-8. Evaluation of surge tank gas release	501
XVI-3-1. Composite plot of TREAT experimental data, aluminum-water reaction	508
XVI-3-2. ATR aluminum-water reaction	510
XVI-3-3. Aluminum-steam reactions, levitation method (8 mm sphere)	512
XVI-3-4. Effect of energy release on cylindrical vessel	515
XVI-3-5. Effect of energy release on spherical vessel	516

XVII-4-1. Atmospheric dilution factors based on Sutton's equation	534
XVII-4-2. Direct gamma dose rate at the MTR following a fuel transfer accident	536
XVII-4-3. Direct gamma dose rate at the ETR following a fuel transfer accident	537
XVII-4-4. Direct gamma dose rate at the evacuation area following a fuel transfer accident	538
XVII-4-5. Beta-gamma immersion dose rates from the plume in the MTR-ETR-ATR area following a fuel transfer accident	539
XVII-4-6. Beta-gamma immersion dose rates from the plume in the MTR-ETR-ATR area following a fuel transfer accident	539
XVII-4-7. Iodine ingestion dose rates to the thyroid in the MTR-ETR-ATR area following a fuel transfer accident	539
XVII-4-8. Iodine ingestion dose rates to the thyroid in the MTR-ETR-ATR area following a fuel transfer accident	539
XVII-4-9. Direct gamma dose rate at MTR following a complete core meltdown	541
XVII-4-10. Direct gamma dose rate at ETR following a complete core meltdown	542
XVII-4-11. Direct gamma dose rate at evacuation area following a complete core meltdown	543
XVII-4-12. Direct gamma dose rate at one mile from ATR following complete core meltdown	544
XVII-4-13. Beta-gamma immersion dose rates from the plume in the MTR-ETR-ATR area following a complete core meltdown	545
XVII-4-14. Beta-gamma immersion dose rates from the plume in the MTR-ETR-ATR area following a complete core meltdown	545
XVII-4-15. Iodine ingestion dose rates to the thyroid in the MTR-ETR-ATR area following a complete core meltdown	545
XVII-4-16. Iodine ingestion dose rates to the thyroid in the MTR-ETR-ATR area following a complete core meltdown	546
XVII-4-17. Beta-gamma immersion dose rates from the plume at distances up to 15 miles from the ATR following a fuel transfer accident	548

XVII-4-18. Iodine ingestion dose rates to the thyroid at distances up to 15 miles from the ATR following a fuel transfer accident	548
XVII-4-19. Beta-gamma immersion dose rates from the plume at distances up to 15 miles from the ATR following a complete core meltdown	549
XVII-4-20. Iodine ingestion dose rates to the thyroid at distances up to 15 miles from the ATR following a complete core meltdown	549
XVII-4-21. Integrated iodine dose to the thyroid	551
XVII-4-22. Integrated beta-gamma immersion dose	552
A-1. Irradiation-induced growth of beryllium	565

TABLES

I-2-1. Design Characteristics	7
I-2-2. Expected Performance Characteristics	11
II-1-1. Population Distribution Around NRTS -- 1960 Census	23
II-1-2. Personnel at NRTS Areas -- June 1961	26
III-2-1. Number of Penetrations in Gastight Area	72
IV-1-1. Minimum Tensile Properties -- Room Temperature	95
IV-1-2. Reactor Vessel Nozzles and Penetrations	96
IV-1-3. ASME Code Stress Analysis Summary	100
IV-1-4. ATR Vessel Nozzle Design	100
IV-2-1. Summary of Calculated Dose Rates	114
IV-2-2. Radiation Control Areas	115
IV-3-1. Pressure Tube Design Conditions	121
IV-3-2. In-Pile Pressure Tube Assembly Dimensions	121
V-1-1. Fuel Element Materials	143
V-1-2. Coolant Channel Deviations	145
V-1-3. Hydraulic Buckling Tests Results	148
V-1-4. Destructive Proof-Tests of ATR Fuel Elements	151
V-1-5. Thermal and Pressure Azimuthal Stress and Plate Deflection	155
V-1-6. Axial Thermal Stress Distribution Plate 19 Peak Power Element at 25 Inches Below Top of Plates	157
V-1-7. Variation in Radial Clearance Between Control Cylinder and Reflector at Core Midplane	164
V-1-8. Hafnium Tensile Properties	166
V-2-1. Reactivity Distribution and Movable Control	168
V-2-2. Reference Core Conditions	169
V-2-3. ATRC Safety Rod Worths -- Dollars	171
V-2-4. Beginning-of-Life Total Temperature Coefficients	172

V-2-5. Beginning-of-Life Void Coefficients	173
V-2-6. ATR Experiment Performance as a Function of Lifetime . . .	176
V-2-7. Values of ℓ^*/β From Pulsed Neutron Experiments in the ATRCE	179
V-2-8. Control Rod and Mechanism Data	181
V-2-9. Bare Reactor Threshold Lengths	186
V-3-1. ATR Thermal Hydraulic Parameters	194
V-3-2. Flow Summary	195
V-3-3. Summary of Wall Temperature Maximizing Factors	197
V-3-4. Error Propagation Data	200
V-3-5. Thermal Burnout Summary	202
V-3-6. Results of Hot Channel Error Propagation Analyses	203
VI-1-1. Primary Coolant System Operating Conditions	211
VII-2-1. Effect of Calibration on Instrument Reading	248
VII-3-1. Test Blocks	258
VII-4-1. Nuclear Control and Safety Actions	265
VII-4-2. Process Control and Safety Actions	266
VII-4-3. Control and Safety Actions for Low Power Runs	268
VII-4-4. Experiment Control and Safety Actions	269
VII-4-5. Miscellaneous Control and Safety Actions	270
VII-6-1. Location of Beta-Gamma Monitors.	291
VIII-1-1. Operating Characteristics of the Safety Rod Drives	312
VIII-1-2. Life Test Results	319
VIII-1-3. Life Tests Scram Times	319
VIII-2-1. Neck Shim Rod Cable Drive Characteristics	326
VIII-2-2. Neck Regulating Rod Shim Drive Characteristics	326
VIII-2-3. Neck Regulating Rod Servo Drive Characteristics	327
VIII-3-1. Operating Characteristics of the Outer Shim Control Drives .	337

VIII-3-2. Construction Materials of Outer Shim Control Drive	337
VIII-3-3. Mechanism Backlash	341
IX-1-1. Electrical Distribution Voltages	346
XIII-1-1. Transportation Plan	414
XVI-2-1. Maximum Allowable External Leak Size -- Pressure Limited.	485
XVI-2-2. Maximum Allowable External Leak Size -- Flow Limited . . .	487
XVI-2-3. Top Shield Displacement	497
XVII-3-1. Timetable for Postulated Complete Core Meltdown	529
XVII-4-1. Meteorological Parameters	534
XVII-4-2. Maximum Integrated Dose Received by Guard During Evacuation After a Fuel Transfer Accident	540
XVII-4-3. Integrated Dose Received by the Guard During Evacuation Following Complete Core Meltdown	547
XVII-5-1. Elapsed Time in Days to Initial Reentry After ATR Accidents	553
XVII-5-2. Minimum Time to Evacuate Following a Complete Core Meltdown at ATR, Cases A and A-S Weather	554
XVII-5-3. Time to Resume Continuous Operation Following an Accident at ATR -- Days.	555
C-I. ATR Fission Product Inventory After 17 Days Operation	583
D-I. Reactor Building Leak Rate Computations	593
F-I. Error Propagation Analysis for Thermal Burnout for Nominal ATR Full Power Conditions	609

VIII. CONTROL ROD DRIVES

VIII-1 Safety Rod Drives

VIII-2 Neck Shim and Regulating Rod Drive Mechanisms

VIII-3 Outer Shim Control Drive Mechanism

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

VIII-1. SAFETY ROD DRIVES

The ATR control elements are moved by four different drive systems. The four drive systems provide for controlled motion of the safety rods, outer shim control cylinders, neck shim rods, and regulating rods. The general arrangement of the rod drive systems is shown in Figure VIII-1-1.

1. GENERAL DESCRIPTION

The safety rod drive mechanism consists of three major assemblies: the vertical-rod actuator assembly, the horizontal drive-shaft assembly, and the drive package assembly. See Figures VIII-1-1 and VIII-1-2. The vertical-rod actuator assembly consists of two sections. One is the snubber assembly which is stationary and the other is the rack tube which transmits motion to and positions the safety rod. The inner and outer snubber tubes provide an annulus for the rack tube. The rack tube, connected to the safety rod by a threaded swivel joint, is aligned within the outer snubber tube by upper and lower guide bushings. Several discharge orifices are located in the snubber tubes to allow reactor cooling water to escape at a controlled rate from the snubber chamber. When the control rod and rack tube are cocked in the up position, water is discharged at a high rate through the orifices to assure maximum rod acceleration for the first 12 inches of travel following a scram. The final phase of rod travel is controlled by smaller restrictive orifices designed to decelerate the safety rod and attached components.

The horizontal drive-shaft assembly transmits power from the drive package to raise the safety rod to its full out position. A pinion gear, located at the end of the horizontal drive shaft, is meshed to the rack tube through an opening in the outer snubber tube. This horizontal assembly uses a flexible coupling which compensates for initial installation misalignment and thermal distortions. The horizontal drive-shaft assembly is supported by ball bearings, and is housed in a guide tube penetrating the reactor vessel through a flange. A rotary face seal, attached to the flange, provides a seal surface to contain the primary coolant.

The drive package assembly, located outside the reactor vessel, is attached to the horizontal guide tube. The assembly consists of a gear motor, an accelerating (scram) spring, electro-magnetic clutch, position indicating and control equipment. The gear motor transmits power to the horizontal drive-shaft assembly through a pair of spur gears via the electro-magnetic clutch. The gear motor provides rotational torque to the horizontal drive-shaft raising the rack tube and safety rod to the full up (cocked) position and at the same time compressing the scram spring. A scram signal interrupts power to the electro-magnetic clutch coil causing the clutch plates to separate, uncoupling the horizontal drive-shaft from the drive motor. The safety rod is accelerated downward by the combined hydraulic, scram spring, and gravitational forces acting on the mechanism.

① CAPSULE TRENCH	⑪ REACTOR VESSEL
② PRESSURE TUBE ASSEMBLY	⑫ NECK SHIM ROD HOUSING
③ OUTER SHIM GEAR BOX	⑬ CORE REFLECTOR TANK
④ OUTER SHIM CONTROL DRIVE	⑭ CORE
⑤ OUTER SHIM DRIVE SERVICE AREA	⑮ NECK SHIM ROD NOZZLE EXTENSION
⑥ SAFETY ROD DRIVE SERVICE AREA	⑯ NECK SHIM REGULATING ROD DRIVE UNIT
⑦ SAFETY ROD DRIVE	⑰ WORKING PLATFORM
⑧ SUBPILE ROOM	⑱ NECK SHIM ROD DRIVE ACCESS ROOM
⑨ SHIELDED LOOP TRAP ASSEMBLY	
⑩ NECK SHIM CONTROL DRIVE UNIT	

FIG. VIII-1-1 CONTROL ROD DRIVE ARRANGEMENT.

① FLUX TRAP BAFFLE	⑪ MAINTENANCE SEAL (INFLATABLE)	⑳ CORE SUPPORT TANK
② SAFETY ROD GUIDE TUBE	⑫ DRIVE MOTOR (INTEGRAL GEAR BOX)	㉑ PRESSURE TUBE ASSEMBLY
③ HAFNIUM POISON INSERT	⑬ BALL NUT SCRAM SPRING ASSEMBLY	㉒ SAFETY ROD POISON INSERT
④ INSULATING JACKET	⑭ LIMIT SWITCHES	㉓ FLUX TRAP FILLER
⑤ PRESSURE TUBE	⑮ POSITION INDICATING TRANSMITTER	㉔ SAFETY ROD HOUSING
⑥ SUPPORT BEARINGS	⑯ MAGNETIC CLUTCH	㉕ DRIVE PINION
⑦ DRIVE SHAFT	⑰ BIOLOGICAL SHIELD	㉖ BOTTOM HEAD CLOSURE PLATE
⑧ FACE SEAL	⑱ REACTOR VESSEL	㉗ PRESSURE TUBE GLAND SEAL
⑨ DRIVE NOZZLE	㉙ RIGID COUPLING	㉚ PROXIMITY SWITCH LEAD
⑩ SEAL WATER SUPPLY	㉛ FLEXIBLE COUPLING	㉜ HOUSING SEAL

FIG. VIII-1-2 SAFETY ROD DRIVE ARRANGEMENT.

1.1 Location and Arrangement

Each of the five safety rods is equipped with its individual drive assembly. The location of safety rods and drives can be changed to other flux trap positions with the exception of the center flux trap. Additional restrictions on location of safety rods are imposed by particular experiment arrangements such as the gas loop design.

The drive shafts and guide tubes extend radially in pairs from the vessel through the biological concrete shield into the safety rod drive service area. The safety rod service area contains the drive package assemblies.

1.2 Operating Characteristics

The design operating characteristics of the safety rod drives are summarized in Table VIII-1-1.

TABLE VIII-1-1

OPERATING CHARACTERISTICS OF THE SAFETY ROD DRIVES

Stroke	36 inches
Bite (Normal poison position with rods cocked)	3 inches inserted
Withdrawal Rate -- 4 rods	4.5 inches/minute
Withdrawal Rate -- 1 rod	9.0 inches/minute
Scram Release Time	less than 25 msec
Scram Time (Including release time)	138 msec -- first 12 inches of travel
Scram Time (Open-head operation)	280 msec -- first 12 inches of travel
Life Expectancy	10 years

1.3 Materials of Construction

The following materials were selected for critical safety rod drive components:

Component	Material
Safety Rod Housing	17-4 PH, H1100, Stainless Steel
Rack Tube	17-4 PH, H1100, Stainless Steel
Horizontal Drive-Shaft Pinion Gear	17-4 PH, H1100, Stainless Steel
Pinion Gear Shaft	17-4 PH, H1100, Stainless Steel

<u>Component</u>	<u>Material</u>
Horizontal Drive-Shaft Ball Bearing	Stellite Number 19 Races, Stellite Number 3 Balls
Horizontal Drive Shaft	304 L Stainless Steel
Rotary Face Seal	Graphitar Grade 67 on Stellite Number 3
Rack Tube Guide Bushings	Stellite Number 3

The selection of the materials of construction were based on design requirements and performance characteristics during the test program.

The major material changes that resulted from performance tests were replacing the 17-4 pH stainless steel guide bushings with stellite number 3 bushings and changing ball bearing material. The bushings were also modified mechanically.

1.4 Access

The safety rod drive equipment located in the service area for ease of access includes: drive motor, gear box, magnetic clutch release mechanism, scram spring, primary water seal, position indicator transmitter, and limit switches.

The hafnium poison sections and flux trap filler pieces, located on the safety rod, are accessible from the reactor top-head area. The safety rod rack tube assembly is the only moving part of the complete assembly which cannot be removed without replacing the experiment pressure tube assembly.

The horizontal drive-shaft assembly can be removed and replaced through the safety rod nozzle opening by using a valved lock system. The drive shaft splines incorporate a double tooth -- double space matching fit configuration to ensure proper indexing during reassembly. Access to the drive units is limited to reactor shutdown periods.

1.5 Seals

The reactor primary water seal system is a rotary face seal. Approximately 5 gpm demineralized water is introduced at the rotary seal to prevent reactor primary coolant from contacting the seal assembly. A pressure actuated backup seal in the bearing housing limits gland seal water leakage into the drive cubicle in the event the rotary seal should fail. An inflatable seal is incorporated into the seal assembly for use during maintenance operations to service the rotary seal and to facilitate horizontal drive shaft removal. This seal is actuated by an air supply. The air supply contains a shutoff and bleeder valve. The bleeder valve is open during operation to ensure total relaxation of the seal.

1.6 Cooling

The safety rod assembly is cooled by reactor coolant which flows downward through the several concentric annuli between the experiment insulating jacket, the guide tube, guide tube poison and filler inserts, and the flux trap baffle. The core pressure drop (100 psi) provides the driving force for adequate cooling

flow to remove heat primarily from gamma-ray absorption. Below the core the coolant passes through orifice holes in the rack tube and safety rod housing into the flux trap plenum and finally to the flow distribution tank.

1.7 Drive System

The gearmotor is a self-locking double worm reduction unit having an output torque capacity of 3400 ± 200 inch-pounds. The gearmotor will normally work against the hydraulic force, the safety rod and rack tube assembly weight, the fluid frictional drag forces, the friction forces, and the scram spring force.

The unit is provided with a torque limiting device to limit the torque to 2800 inch-pounds. This torque is reached in the event the full-out indicator switch fails to operate during rod withdrawal.

The electro-magnetic clutch provides quick release action following a scram signal. The clutch is set to slip at 3200 inch-pounds torque to provide overload protection in the event both limit switch and torque limiting device fail. The clutch is a commercially available unit of conventional design (Maxi-torq series 9000, model 800).

The scram spring is coupled through a ball screw assembly to the drive shaft. This spring is compressed by the drive shaft as the safety rods are withdrawn. The resulting torque is always in the direction of rod insertion.

1.8 Position Indication

A continuous position indication is provided by a position transmitter connected directly to the horizontal shaft. Electric limit switches are used for full up and down indication and for control interlocks. Each drive has two electric upper and two electric lower limit switches. One switch in each position is a spare. An independent backup indication of the full-in position is provided by a pair of hermetically sealed proximity-type indicators (one a spare) located in the snubber section of the safety rod housing.

Normally the upper limit switch will prevent over travel of the safety rod. In the event of switch failure, a positive mechanical stop in the drive gear housing is provided.

1.9 Scram Action

The safety rod, after clutch release by a scram signal, is accelerated downward by the following forces:

- (1) The fraction of core pressure drop (100 psi) used in overcoming hydraulic drag and entrance losses caused by coolant flow through the mechanism.
- (2) The hydraulic weight of the vertical assembly -- 160 pounds.
- (3) The scram spring torque -- 640 inch-pounds.

Open-head (for flux mapping) scram action will not have hydraulic forces available for rod acceleration. Figure VIII-1-3 shows a typical time-displacement curve for normal rod scram performance.

FIG. VIII-1-3 ATR SAFETY ROD SCRAM PERFORMANCE CURVE.

2. DESIGN CONSIDERATIONS

The safety drive mechanism has been designed to prevent seizing, galling, corrosion, and wear. Design considerations for critical components are presented below.

2.1 Rack and Pinion

Design considerations for the rack and pinion are strength, wear, corrosion, and shock resistance. The use of a high contact ratio permitted greater load carrying capacity. By increasing the ratio of the arc of action to the circular pitch, more teeth are brought into load sharing position thereby reducing the load on any one tooth. The calculated nominal contact ratio of 2.3 distributes the load between 2 to 3 pairs of teeth when pitch lines coincide. Based on a maximum backlash of 0.019 inch caused by cumulative tolerances and clearances in the vertical driveline, the contact during one rotational cycle is

two pair contact -- 68 percent of the cycle

one pair contact -- 32 percent of the cycle.

In addition to the stresses in the teeth caused by nominal loading, two additional factors were considered: impact loading and induced vibration. Based on calculations which include stress concentration factors and prototype test results, the rack and pinion teeth designs are satisfactory.

2.2 Outer Snubber Tube

The critical factors determining design of the outer snubber tube are anti-galling properties, wear, and corrosion. Since sliding contact between the outer snubber tube and the rack tube guide bushings is required, the contact materials must also exhibit compatibility. The snubber tube material selected was 17-4 PH stainless steel precipitation hardend (H1100). The use of stellite number 3 guide bushings as mating surface with the snubber tube has provided a compatible system to minimize galling.

2.3 Rack Tube Guide Bushings

Prototype testing required changing the bushing material from 17-4 PH stainless steel to stellite number 3 because of excessive galling. The guide bushing design provides diametral clearances within the outer snubber tube of 0.013 inch maximum and 0.010 inch minimum. These clearances provide adequate buffing action and prevent excessive crud buildup. A flow of filtered water has also been provided to minimize crud accumulation in the lower guide bushing area.

2.4 Horizontal Shaft Ball Bearings

Calculated maximum dynamic compressive stress for the preloaded stellite number 3 pinion bearings is 306,000 psi compared to a design value of 412,000 psi. Ball separators have been designed for maximum water flow through the bearings. Seal water is used to flush the bearings continuously to prevent crud accumulation.

2.5 Scram Spring

The scram spring has been designed to limit stresses to 72,000 psi. The design has minimized the possibility of spring buckling to prevent undue lateral loads on the sliding members. A multiple fracture of the spring coils has been calculated to increase the scram time 40 to 70 msec but would not prevent a scram.

The safety rod spring has been subjected to 100 percent liquid penetrant or magnetic particle inspection.

2.6 Ball Nut and Screw Assembly

A static load of 2000 pounds (maximum) is normally applied by the scram spring to the translating ball nut. This force acts to rotate the screw in the direction of rod insertion when the clutch is de-energized. The design of this assembly considered shock resistance and useful life and is adequate for the loads.

2.7 Electric Clutch

The design of the electric clutch considered the holding requirements and the magnetic and mechanical forces affecting the release and breakaway time. Several modifications were made to improve the clutch characteristics. The

modifications included improving the magnetic flux shape by adding an RC circuit, slotting the clutch coil housing and plates, and optimizing the air gap. Modifications to reduce mechanical drag during breakaway include improving clutch disc finish, shape, and spring action.

2.8 Rotary Seal

The rotary face seal is relatively maintenance free and is inherently free of shaft binding characteristics. The measured amount of wear on the seal face was 0.0024 inch after extensive life testing.

2.9 Misalignment of Components

The design of the safety rod and drive assemblies considered interference between the experiment pressure tube assembly, the safety rod, and flux trap baffle (guide tube). Excessive misalignment of the reactor nozzles as a result of improper fabrication and thermal conditions was also considered.

The vertical assembly incorporates a swivel joint which permits the maximum misalignment dictated by the clearance between the safety rod proper and the inside diameter of the flux trap baffle (0.060 inch).

The horizontal driveline contains a flexible coupling which has an angular misalignment capability of $+0^{\circ}$ -15' at the design torque rating. This allows a maximum misalignment of 0.5 inch measured at the center of the nozzle outer flange in relation to the true axis of the horizontal nozzle.

3. COMPONENT MALFUNCTION

Although the design and testing program of the ATR safety rods minimize the probability of failure and offer evidence of satisfactory performance, malfunction of several components can lead to slow scram time or scram failure. These components are:

Outer snubber tube and rack mounted guide bushings

Horizontal shaft ball bearings

Rack and pinion

Ball nut and screw assembly

Electric clutch.

Crud buildup in critical areas of the first three of the above listed components represents the most probable malfunction. Maximum flushing is available for the outer snubber tube and bushings when the rod is in the cocked position. Crud buildup would tend to occur on edges opposite the leading edges of sliding members. This would probably lead to slow scram time from wedging action rather than scram failure. The horizontal bearings are flushed with demineralized seal water preventing accumulation of any primary coolant crud.

Electric clutch disc wear can cause slow release and breakaway time. The discs are encased preventing exposure to contamination.

The ball nut and screw assembly and scram spring are enclosed to prevent dirt contamination.

Periodic rod drop tests and preventative maintenance procedures will detect gradual buildup of crud and mechanical problems in critical drive areas.

It is concluded that malfunction of one safety rod drive system is possible because of crud buildup or mechanical failure. Multiple simultaneous failure of several drive systems is not considered credible.

4. PROTOTYPE SAFETY ROD DRIVE ASSEMBLY TESTS

A prototype ATR safety rod drive mechanism was subjected to three series of tests to determine if the drive performance was satisfactory. The tests consisted of:

(1) Operational Testing

The purpose of these tests was to ensure that operating design conditions were achieved. Various modifications were made during the tests to achieve performance.

(2) Life Testing

These tests demonstrated reliability under simulated reactor operating conditions.

(3) Miscellaneous Testing

These tests included:

- (a) misalignment tests
- (b) crud tests
- (c) inflatable seal tests
- (d) relay panel tests
- (e) scram tests without filler plates
- (f) service tool tests

Operating conditions during the tests simulated reactor conditions with respect to hydraulics and temperature. The coolant chemistry was not controlled during all tests with respect to pH and oxygen content.

4.1 Operational Testing

Operational testing was performed to examine each drive component performance separately. Modifications made to the mechanism as a result of these tests include:

Lengthening the scram spring from 26 to 27 inches was done to provide additional acceleration.

Changes were made in clutch design.

Horizontal shaft bearing retainer was modified and ball bearing material changed.

The guide bushings were modified and the material changed.

The snubber tube orifice sizes were changed.

Additional minor changes were made to various components during the tests to improve the overall mechanism performance.

4.2 Life Tests

These tests were designed to demonstrate the ability of the safety rod drive mechanism to scram after repeated cycling. Coolant temperature was varied during certain tests and 10 scrams were performed at zero flow. The tests were run in three phases and results are shown in Table VIII-1-2.

TABLE VIII-1-2
LIFE TEST RESULTS

Number Scram Cycles	Average Scram Time--msec (12 inches of travel)	Maximum Scram Time--msec (12 inches of travel)
200	138	140
200	141	149
1100	144	151
10 (zero flow)	273	285

Two additional tests were run allowing the rod to remain cocked for periods of 100 and 300 hours, respectively, before performing 20 scrams. The scram times are shown in Table VIII-1-3.

4.3 Miscellaneous Tests

4.31 Misalignment Test. Horizontal angular misalignment limit as designed was 0° 10'. Maximum vertical misalignment between pressure tube assembly and flux trap baffle tube, based on drawing tolerances, was 0.045 inch. A test was run using 0° 15' horizontal

TABLE VIII-1-3
LIFE TESTS SCRAM TIMES

Period of Test -- Hours (rods cocked)	Average Scram * Time Before Test (12 inches rod travel)
100	138
300	136
Average (20 cycles) * Scram Time After Test (12 inches rod travel)	First Scram * Time After Test (12 inches rod travel)
140	139
137	137
* Scram Time in msec	

angular misalignment and 0.060 inch vertical misalignment. Results of the test showed scram time (12 inches rod travel) of 138 msec before and after misalignment.

4.32 Crud Tests. A crud test was performed to simulate introduction of both probable and accidental concentrations of crud into the reactor. An examination of accidental crud introduction in the ETR was made. Based on this study, a crud composite consisting of 75 grams of cation resin, 400 grams of magnetite, 15 grams sand, and one square inch of blotting paper was prepared. The rod was cocked and all seal and flush water lines operating prior to circulating any crud. The test was performed in the following steps:

- (1) 4.2 grams of the crud mixture were circulated in one loop water volume for 30 minutes. The rod was then scrammed. This procedure was repeated five times.
- (2) 100 grams of the crud mixture were added, circulated, and step (1) repeated.
- (3) 263 grams of the crud mixture were added, circulated, and step (1) repeated. The above crud additions were cumulative for each step.

Results indicated an average scram time of 145 msec for all tests compared to 148 msec prior to adding crud. Figures VIII-1-4 and VIII-1-5 show crud accumulation in the outer snubber tube and front pinion bearing following disassembly. A previous crud test resulted in a jammed mechanism. This test is not detailed because of improper test procedures, possible mistake in resin type used, and an unrealistic composition.

4.33 Inflatable Seal Test. This test measured leak rate simulating a failure of the inflatable seal during maintenance operation.

With the test system pressurized to simulate reactor hydrostatic head, the inflatable seal was pressurized with air and the rotary face seal removed. No leakage occurred. Air pressure to the inflatable seal was reduced to atmospheric pressure and the leak rate was found to be 9 gpm.

4.34 Relay Panel Tests. The purpose of this test was to determine delay time following a relay scram. A circuit simulating the reactor system was included in the test system. Results showed a 10 millisecond scram delay time caused by the relay circuitry.

4.35 Scram Tests Without Filler Pieces. A series of scram tests was made to determine the effect of removing the aluminum filler pieces from the safety rod. Results indicated no significant effect in removing the filler pieces.

4.36 Service Tool Test. This test was performed to determine the performance of the service tool in removal and replacement of the entire horizontal drive assembly. A complete disassembly and assembly of the horizontal drive system was successfully performed under conditions simulating reactor hydrostatic head.

FIG, VIII-1-4 DEPOSITS IN OUTER SNUBBER TUBE AFTER CRUD TEST.

5. SUMMARY

FIG. VIII-1-5 DEPOSITS ON FRONT PINION BEARING AFTER CRUD TEST.

The ATR safety rod drive mechanism has performed satisfactorily during extensive testing in an environment similar to ATR conditions. Scram time was not significantly effected by crud conditions simulating both probable and accidental concentrations expected in the ATR. Misalignment tests have shown the mechanism capable of satisfactory performance under conditions more severe than expected from maximum tolerance stackup. Wear and corrosion of the production components were satisfactory.

The five production mechanisms were subjected to vendor testing prior to site shipment. The safety rod drive mechanisms, following reactor installation, will be tested under both operation and open-head hydraulic conditions prior to loading fuel in the reactor.

Based on design calculations performance testing, and expected results of preoperational check-out, it is concluded that the ATR safety rod drive will meet required reactivity insertion rates under conditions of normal operation, misalignment stackup, probable and accidental crud concentrations introduced to the reactor coolant, and all credible malfunctions that can occur.

VIII-2. NECK SHIM AND REGULATING ROD DRIVE MECHANISMS

1. GENERAL DESCRIPTION

The neck shim and regulating rods are slender, fluted hollow poison rods approximately 48 inches long. The rods are vertically oriented and arranged in four groups of six along the centerline of each neck of the neck shim housing. The first, second, third, fifth, and sixth rods numbered radially outward from the center in each neck are used for shim control. The fourth position contains a rod that can be used for either shim or regulating control.

Each neck shim rod has a nonpoison follower section attached to its lower end. Rod withdrawal is accomplished by driving the rod upward with a flexible cable system designed for push-pull motion. The cable is actuated by a mechanism located below and external to the reactor vessel. The rods are housed in guide tubes located in the neck shim housing. The drive cable is enclosed in tubing (conduit) acting as cable guide and support to prevent buckling under compressive loading as shown in Figure VIII-2-1.

Each regulating rod operates within a guide tube and has no follower section. Withdrawal is accomplished by driving the rod downward out of the active core region. The regulating rod is rigidly attached at its lower end to a solid drive rod extending down through a pressure envelope below the reactor bottom head. The drive rod penetrates the pressure envelope through a labyrinth-type seal. The general arrangement is shown in Figure VIII-2-1.

Each of the four neck shim rod drive assemblies consisting of five shim and one regulating rod drive and their associated drive cables, conduit, drive rods, and guide tubes is brought out of the reactor through one of four pressure housing sections.

Each pressure housing section consists of an upper extension assembly, pressure housing assembly, and lower extension assembly. The upper extension assembly contains a transition section at the top mating with the bottom of a neck region of the neck shim housing by means of dowel pins. The in-line arrangement is changed in the transition section to a tubular geometry containing the five shim rod drive cables and conduit arranged concentrically around the regulating rod drive guide tube. The tubular upper extension assembly bottom penetrates several inches into the pressure housing assembly below the reactor bottom closure plate. The pressure housing assembly serves to form a reactor coolant water-tight seal. It consists of a one piece, tubular structure welded at the top to a nozzle extension of the reactor bottom closure plate. The tube extends downward through the subpile room to a point just below the rod access room ceiling. The lower extension assembly is a tubular structure similar to the upper extension assembly and is located inside and sealed to the pressure housing. The assembly contains the drives in the same concentric pattern described above. The lower end of the assembly has a closure cap and seal. Tube fittings welded concentrically to the cap provide the means of cable exit. The centrally located regulating drive rod exits through the seal assembly. Tubing (conduit) routes the cables from each fitting to its respective drive actuator assembly.

FIG. VIII-2-1 ATR NECK SHIM ROD DRIVES.

The shim rod drive actuating assemblies (one per rod) are wall mounted around the rod access room. Each actuating assembly consists of a gear motor, torque limiting coupling, cable drive housing assembly, cable well section, and associated position and limit switch instrumentation.

The gear motor drives a shaft that penetrates the cable drive housing assembly through a chevron packing seal assembly. A hobbled drive wheel, attached to the shaft, imparts the drive power to the flexible cable. The flexible cable is helically wound wire surrounding a wire core mating with the hobbled drive wheel in the same manner as a rack and pinion arrangement. Conduit containing the flexible cable is attached to the housing through pressure tight fittings from the pressure housing and the cable well sections. The cable is therefore in contact with pressurized water throughout its length. The cable well contains the free end of the cable and serves as a pressure vessel. The bottom end of the cable well contains a magnetic reed switch assembly to give positive indication of cable bottoming and provide a mechanical stop for the cable. A position transmitter and limit switch arrangement provides a signal proportional to the rod position.

The regulating rod drive actuator assemblies are located directly under each of the four pressure housing sections on the subfloor of the rod access room. Each actuator assembly is vertically orientated to the centerline of the rod guide tube. The assembly consists of a main frame subassembly, drive unit, position transmitter assembly, limit switch assemblies, and a buffer assembly. Two of the units are equipped with a servo mode of operation and two with a shim mode of operation.

The main frame subassembly is an L-shaped cast iron member to which the drive unit is rigidly attached. It provides the alignment and means of support for the drive rack attached to the drive rod. It also serves as the mount for all components of the drive.

Each of the two servo drive units consists of a servomotor and a tachometer arranged in a closed servo loop. The motion of the servomotor is transmitted to the regulating rod via a rack and pinion arrangement, the pinion being on the end of the servomotor shaft. The tachometer produces electrical signals for the servo mode of operation.

Each of the two shim drive units consists of a gear motor with an overload coupling attached to the shaft. A pinion gear at the end of the shaft engages the drive rack. The drive rack is moved at constant speed up or down.

The position transmitter consists of a selsyn unit geared to the drive rack for the servo drive unit. The shim driven transmitter is a shaft position encoder using an electronic unit, called a gate array, geared to the drive rack.

Two limit switch assemblies enclose the drive rack up to the seal on the lower extension assembly. A total of seven roller actuated switches is mounted at different intervals. The switches are actuated by the drive rack.

The buffer and buffer subassembly consist of two similar hydraulic telescopic-type buffers separated by a support tube. The buffer is attached to the bottom face of the frame subassembly and extends vertically downward. A striker assembly is attached to the bottom end of the drive rack and contacts the piston of each buffer. This limits the stroke of the regulating rod vertically

in the event of failure of the dynamic braking of the servomotor. In addition, the mechanical stops of the regulating rod stroke are provided by the buffer and buffer subassembly.

1.1 General Characteristics of the Neck Rods

Neck shim rod cable drive characteristics are listed in Table VIII-2-1. The neck regulating rod shim drive characteristics are listed in Table VIII-2-2. Neck regulating rod servo drive characteristics are listed in Table VIII-2-3.

TABLE VIII-2-1

NECK SHIM ROD CABLE DRIVE CHARACTERISTICS

Design Life	
Basic	10 years
Minimum Maintenance-Free Period	1 year
Speed	6.0 in./min
Stroke	48 in.
Gear Motor	
Rated Torque	190 in.-lb
Type	3 phase induction
Rated Capacity	1/20 hp
Voltage	208 Volts ac
Position Indication System	
Overall Accuracy	± 1 in.
Seals	
Static	O-ring, Buna-N
Dynamic (drive shaft)	Chevron, Buna-N nitrile spring loaded
Purge Water Inlet Pressure	385 psig

TABLE VIII-2-2

NECK REGULATING ROD SHIM DRIVE CHARACTERISTICS

Design Life	10 years (gear motor components)
Speed	6.0 in./min
Gear Motor	
Rated Torque	325 in.-lb
Type	3 phase induction
Rated Capacity	1/20 hp
Voltage	208 Volts ac
Position Indication System	
Overall Accuracy	± 1 in.

1.2 Materials of Construction

A list of materials of construction for the neck rod assemblies is shown below:

Neck shim rod guide tube	-- Aluminum
Neck rod poison section	-- Fluted hollow hafnium rod
Neck Shim rod follower	-- Aluminum
Neck shim rod flexible cable	-- H and E Aircraft type 321 stainless steel helically wound cable
Cable conduit	-- Inconel 600
Regulating rod guide tube	-- Inconel -- X
Regulating drive rod	-- Inconel -- X (Section penetrating seal is chrome plated).

TABLE VIII-2-3

NECK REGULATING ROD SERVO DRIVE CHARACTERISTICS

Design Life	1 year
Stroke	48 in.
Speed	50 in./sec maximum
Acceleration	5-45 in./sec in 0.04 sec
Response	to 12 cps $\pm$ 3 dB
Servo Motor Data	
Type	dc permanent magnet
Rated Capacity	1 hp @ 1140 rpm
Maximum Speed	1500 rpm
Integral Brake Torque (static)	6.0 ft-lb
Tachometer Generator	
Type	dc permanent magnet
Ripple	1 percent maximum @ 500 rpm 4 percent maximum @ 25 rpm
Linearity	$\pm$ 1 percent minimum @ 500 rpm and up $\pm$ 4 percent minimum @ 100 rpm and down
Drive Rod Seal	
Type	Labyrinth
Leak-Off	6.0 gph maximum
Purge Water Inlet Pressure	385 psig
Position Indication System	
Overall Accuracy	$\pm$ 1/4 in.
Type	Standard Synchro
Rotation for 48 in. Stroke	353°
Power Supply	115 volts ac
Hydraulic Buffers	
Maximum Design Load	1000 lb
Stroke	3-3/4 in.
Return Time	0.3 second maximum

All materials of construction in contact with reactor coolant have been corrosion tested and found satisfactory with the exception of the type 321 stainless steel flexible cable which has not been tested. Since this component will be replaced periodically along with the control rod, the application appears to be acceptable.

1.3 Access and Maintenance

The actuator assemblies for the neck rods are accessible for contact maintenance. If necessary, complete actuator assemblies may be disconnected and replaced by spare units. The primary maintenance operation to be performed is replacement of the rod and drive cable. This is accomplished by removing the magnetic switch housing, removing the anti-rotation fitting from the cable, and driving the cable upward until free of the drive wheel. Final removal is accomplished inside the reactor vessel. Some leakage (0.1 to 0.2 gpm) results until the magnetic switch housing is reinstalled.

The regulating rod will require periodic removal to replace the poison section. A special clamp tool that attaches to the pressure housing closure cap is provided to hold the drive rod after disconnecting from the rack. The parts of the regulating drive actuator mechanism subject to rapid wear are readily accessible for replacement.

The pressure housing section will not require maintenance except for possible O-ring seal replacement. After removing all cables and drive rod, a plug is placed below the vessel bottom closure plate and above the slip joint between the upper and lower extension assemblies to seal off the opening to the reactor vessel.

1.4 Cooling

Reactor coolant flows downward through the hollow hafnium rod and aluminum follower, and in the two annuli formed by the outside circumference of the rod, the aluminum guide tube, and the neck shim rod housing. The coolant exits through holes in the transition section into the flow distribution tank. The rods and guide tubes contain spacers to ensure that the coolant channels are maintained. Adequate flow is available for cooling gamma heat conservatively calculated to be 64 watts/gram in the hafnium rod.

1.5 Position Indication

1.51 Shim Rod. The shaft position encoder-digital display equipment is used for remote indication of the shim rod position.

Limit switches having arm and roller actuators are provided at each end of the mechanism stroke. The motor driven gear train, which operates the encoder, drives a small rack that actuates the switches.

1.52 Regulating Rod -- Servo. A standard synchro transmitter and receiver is used to read the position of the regulating rod. The servo motor units and synchro transmitters may be placed in any two of four locations, and signal lights representing the four locations are, therefore, provided at each readout on the reactor control panel.

The Servo Drive Data System Position Transmitter consists of a multi-turn wire wound potentiometer coupled to the rear end of the tachometer shaft.

This furnishes the desired signal by applying a regulated voltage supplied from the data processor across the potentiometer.

When a gear motor is installed on a regulating rod assembly (shim control), the synchro position transmitter is replaced with an encoder unit.

Seven limit switches are provided for the regulating rod assembly. These switches provide actuation of the outer shim drum cylinders to maintain the regulating rod in the most desirable location. All switches are heavy duty snap acting type with double-pole double-throw contacts.

1.6 Drive Power Unit

1.61 Servo Regulating Rod. The servo motor power unit provides automatic regulation of the reactor through the regulating rod. The complete regulating drive system is a velocity servo receiving its signal from the nuclear instrumentation. A low inertia dc servo motor, coupled to a tachometer and a 25 turn potentiometer, provides power to move the drive rack and rod in response to signals from the reactor control system. The potentiometer is a position indicating device used to feed a signal to the data processing system. There is a second potentiometer mounted at the synchro transmitter. This potentiometer permits operation of the servo unit manually; it prevents drifting of the unit under manual operation.

The motor frame encloses a spring energized disc brake actuated by the power used to operate the unit. The armature shaft of the motor is double ended and machined on the front end to accept the drive pinion.

The power source for the servo motor affects the manner in which the drive operates.

A rotary dc amplifier rated at 3.0 kilowatts is used as the power source. Suitable electronic amplifiers adapt the reactor instrumentation and control system signals to control the rotary amplifier.

1.62 Shim Rod. The neck shim rod is driven by a 1/20 hp, three-phase, electric motor through a 1800 to 1 double worm reduction gear. No automatic brake or other device is required to stop the motor when the power is cut off because of the high gear reduction and very low shaft output speed. The rated running torque of the gear drive does not exceed the cable load capacity, but the stalled torque does exceed the cable load capacity. Therefore, a torque limiter is provided to prevent possible cable damage.

1.7 Seals

The labyrinth-seal assemblies located at the bottom of the pressure housing section are equipped with demineralized seal water supplies to prevent reactor coolant from contacting the surfaces.

The chevron seals for the shim drive shafts are also equipped with seal water supply lines.

2. DESIGN CONSIDERATIONS

The major design problem associated with the neck shim rods concerned the flexible drive cable and conduit system. The regulating rod design proved to be more than adequate in meeting design requirements.

2.1 Flexible Cable Drive Section

The neck shim rod cable required extensive testing before a satisfactory product was developed. The conduit material and geometry also affected cable performance. The selection of H and E Aircraft type 321 stainless steel helically wound around a stranded core cable was based on test results. Inconel-600 conduit with maximum radius bends tolerated by space limitations yielded satisfactory performance.

An aluminum fitting is swaged to the upper end of the cable to permit a welded connection to the aluminum follower section.

The drive cable passes through a stainless steel guide bushing in the drive housing to support and guide the cable as it passes by the drive wheel.

A stainless steel anti-rotation fitting attached to the end of the cable also acts as a positive stop and contains a permanent magnet which actuates a fully inserted position switch.

The drive wheel rim has special teeth which mesh with the cable helix wire.

The Inconel-600 conduit, housing the flexible drive cable, uses standard flare fittings to provide smooth joints and maintain pressure sealing characteristics.

The aluminum follower section is larger in diameter than the cable conduit. The fully inserted position of the bottom of the rod follower is within one inch of the conduit in the transition section.

2.2 Regulating Rod Drive

The Inconel-X drive rod, is in three sections. The upper two are electro-lized to reduce friction and wear, and the lower section has a chrome plate with a fine finish to operate through a labyrinth seal. The seal assembly consists of floating stellite rings, fitting the drive rod closely, and stellite bushings top and bottom to align the rod as it passes through the seal. Wiper seals on the outside end of the assembly keep dirt out and route leak-off water into the drain connection.

The drive rod, when fully extended below the seal, does not have sufficient column strength to sustain the available load imposed by the drive if a jammed mechanism occurs. Therefore, the limit switch assembly is placed between the seal and drive mechanism and surrounds the drive rod in its extended position. This provides the required support should this type of accident occur.

2.3 Pressure Containing Members

The principal pressure members (the pressure housing and cable wells) are designed in accordance with the current ASME code, Section VIII. All welding and welder qualifications have been performed on a more stringent basis than required by code.

3. COMPONENT MALFUNCTION

An analysis of accidents which could occur to the neck shim rod and regulating rod drive mechanisms is presented.

3.1 Neck Shim Drive Mechanism

3.11 Drive Motor Malfunction. A power loss will stop motor and maintain rod position.

An open phase occurring when the motor is energized will allow the motor to continue to drive until de-energized; if de-energized, it will not drive when re-energized.

Voltage variations between 187 to 229 volts will maintain rod velocity within specifications (5.7 to 6.0 inches/min).

Wiring reversal changes direction of rotation.

3.12 Rod Overtravel. Limit switches have been provided to indicate fully inserted and fully withdrawn positions 0 and 48 inches, respectively. Operating beyond these limits will result in contacting the mechanical stops. Driving against these stops will result in tripping the overload coupling. This is not detrimental to the clutch or mechanism. Driving into the mechanical stops if the overload coupling is locked will cause deformation of the cable helix and the cable drive wheel.

3.13 Position Readout Transmitter. Loss of the position readout transmitter would not affect the ability of the neck shim drive mechanism to move the control rod although the readout equipment would not be operable. Position indication can be obtained by timing from a known location such as actuated by the limit switches (0 and 48 inches) or the magnetic switch.

3.14 Loss of Anti-rotation Device. Should the collet which holds the cable to the anti-rotation block loosen, it is possible for the cable to unscrew from the block. The cable can still be driven, but the indicated position would be in error. If the anti-rotation were completely separated from the cable, the block would fall to the bottom of the cable well and trip the magnetic switch. In no event can the cable lose engagement with the drive wheel; however, there will be an indication that the situation exists by the disparity of the signals between the magnetic switch, the limit switches, and the position readout.

3.15 Stuck Neck Shim Rod or Drive Cable. The neck shim drives are provided with an overload torque coupling which protects the drive from damage; however, no rod motion will take place. Detection of internal binding can be accomplished by monitoring drive motor current and/or power.

3.2 Neck Regulating Drive Mechanism

3.21 Drive Motor. Power loss to system: If power is lost to the system, the amplidyne supplying power to the dc servo motor would start to slow down. This would cause the motor mechanism brake to be engaged. If the motor had been driving in the inserting direction, the regulating rod would decelerate more slowly and could possibly contact the buffer. The mechanism would not be damaged since the velocity of the rod entering the buffer would be equal to or less than the design velocity.

Power loss to motor alone: It is possible to lose power to the motor without engagement of the brake. Under these conditions, it is possible for the gravitational and hydraulic forces acting on the rod to accelerate the rod in the withdrawal direction. This could lead to velocities in excess of the design causing deformation of rack and pinion teeth.

3.22 Tachometer. Failure of the tachometer will prevent the neck regulating drive mechanism from functioning as a servo system.

3.23 Synchro Transmitter. The neck regulating drive will function without a synchro transmitter.

3.24 Overtravel Limit Switches. Failure of these switches will not prevent regulating action.

3.25 Dynamic Brake Limit Switches. Under normal operation these switches reverse the motor so that the drive rack does not contact the buffers. Failure of these switches will allow the rack rod to contact the buffers which stop rod motion without mechanism damage.

3.26 Brake. The brake is used for holding when it is desirable to hold without using the servomotor. The motor is capable of overriding the brake and driving the regulating rod at reduced speed should the brake be engaged inadvertently.

3.27 Thermal Sensor. The thermal sensor signals a motor over temperature condition. Loss of this signal could result in shorting out the motor windings preventing operation of the mechanism. This would result in a situation similar to but not as severe as a loss of power to the motor since some braking action would result.

3.28 Potentiometer. The loss of the potentiometer has no direct effect on the mechanism.

3.29 Buffer Failure. Failure could only occur following a limit switch or motor failure. Under either of these conditions, a buffer failure would stop the rod motion with probable damage to rack teeth.

3.210 Stuck Rod. This condition would cause the servo motor to stall and overheat unless power is discontinued. Overheating will be indicated by a signal from the thermal sensor at the control panel.

3.3 Neck Regulating Shim Drive Mechanism

The drive system and limit switches are identical to the cable driven shim rods, and malfunctions are the same.

4. NECK SHIM AND REGULATING ROD DRIVE PROTOTYPE TESTS

Prototype testing of the ATR neck shim rod and regulating rod mechanisms was performed under conditions simulating reactor environment. A summary of the test results is presented.

4.1 Neck Shim Rod Cable Drive Mechanism

Initial testing of flexible cables showed unsatisfactory performance. Development work including finding new vendor sources was required before a suitable cable was obtained. Environmental life testing has shown the presently designed mechanism has met all specifications with the exception of cable cycle life. The expected life of the cables based on life testing is 300 to 500 cycles. Cable failure is based on frictional effects that cause the torque required to drive the cable to increase beyond the overload limiter release point. This does not present an operational hazard because the gradual nature of the onset of failure permits detection by routine periodic motor power checks. The torque required to drive the cable as a function of cycles is shown in Figure VIII-2-2. The number 2 conduit run provides the most severe conditions of the five types of runs for the cable because of the bending pattern. A total of approximately 2500 cycles were run with H and E cables under ATR environment conditions. Inspection of the various drive mechanism parts indicated satisfactory performance.

FIG. VIII-2-2 NECK SHIM FLEXIBLE CABLE PERFORMANCE CURVES.

4.2 Neck Regulating Rod Mechanism

The ATR prototype neck regulating rod mechanism was subjected to a series of tests under simulated ATR environment. The tests included:

- (1) Component testing
- (2) Buffer tests
- (3) Velocity tests
- (4) Response tests
- (5) Motor heating tests
- (6) Life Tests

The regulating rod was operated for 1000 hours (life test) at 5 cps (maximum amplitude). The rod position was changed during the life test from 6 to 42 inches at a rate of about 18 inches/hour maintaining 5 cps oscillatory motion. The rod was successfully operated for over 1×10^7 cycles. Results of all the tests showed that all design requirements were met.

Performance of the regulating rod shim drive mechanism under environmental and life testing was satisfactory. All design requirements were satisfied. All production components were inspected after testing and found to be free of corrosion and excessive wear.

5. SUMMARY

The ATR neck shim drive and regulating drive mechanisms have performed in a satisfactory manner during extensive testing under environmental conditions simulating the ATR. Although the shim cable drive life is limited to about 300 to 500 cycles, this will probably be greater than the life of the poison section, and replacement can, therefore, be programmed to coincide. Preventative maintenance procedures will easily detect cable problems well in advance of failure. The simultaneous failure of more than one neck shim rod is remote.

Based on the test data, design considerations, and malfunction analysis, the ATR neck shim and regulating rod mechanisms will provide the required reactivity functions in a safe reliable manner.

VIII-3. OUTER SHIM CONTROL DRIVE MECHANISM

1. GENERAL DESCRIPTION

The ATR outer shim control drive mechanism positions the outer shim control element. The element is a curved hafnium plate attached to a 120 degree periphery of a vertical beryllium cylinder which rotates in the reflector section of the reactor core. Sixteen such cylinders are arranged in the beryllium to surround the four outer lobe flux traps. The control cylinders are supported from the drive mechanism. The cylinders are mounted and positioned between two circular aluminum end plates. A full length aluminum shaft through the cylinder is threaded to the end plates. The upper end plate is bolted to a connector column at the bottom of an internal gear box. The beryllium cylinders are free of all loads except those imposed by coolant pressure drop and the cylinder weight in water. The bottom of the full length aluminum rod is restrained laterally by bosses located on the core reflector support tank. There is a total of eight drive mechanisms, each driving two control cylinders. The general arrangement is shown in Figure VIII-3-1.

Each drive mechanism is made up of five major subassemblies. These subassemblies are described as follows:

(1) Outer Gear Box Section

The outer gear box section, located outside the reactor vessel, consists of a drive motor coupled through a mechanical release clutch to a gear box which houses mechanical positive stop guides and a gear train to the electrical position readouts and microswitch actuator stop. A transmitter for the Decitrak digital readout, a potentiometer, and limit switches are also part of this section.

(2) Drive Shaft Inner Housing Section

This section links the outer gear box to a coupling shaft. The housing contains a radial bearing on the inner end and a radial thrust bearing on the outer end positioning the drive shaft. The drive shaft is held in place longitudinally by retainers on the outer end. The reactor coolant primary seal is located toward the outer end of the shaft. A port in the housing allows the introduction of seal water outboard of the seal. A lip seal is provided to confine either seal water or reactor coolant in the event of a seal failure.

(3) Coupling Shaft Section

This section is a spring loaded, two-piece, collapsible unit designed to be installed as the final mechanism connecting link between the drive shaft and inner worm gear box. Torque is transmitted through splines on both pieces of the section, and proper orientation is obtained by missing teeth on the splines coupled to mating sections.

(4) Inner Worm Gear Box

The worm gear box is used to change direction of the mechanism rotational motion from a horizontal input to a vertical output through a 40 to 1 gear reduction. Rotational torque from the coupling shaft turns the double worm shaft engaging the two worm gears. In addition, loads generated by the cylinders are carried by large worm gear bearings in the gear box housing. The double

FIG. VIII-3-1 OUTER SHIM CONTROL DRIVE.

worm shaft is also mounted on bearings. The worm gear box is mounted on a beam assembly secured to the core reflector tank as shown in Figure VIII-3-1.

(5) Shaft Connecting Assembly

The shaft connecting assembly links two cylinders to the worm gears. Rotational loads are carried through splines. The cylinder vertical loads are carried by coupling bolts.

1.1 Operating Characteristics

A list of operating characteristics of the outer shim control drives is shown in Table VIII-3-1.

TABLE VIII-3-1

OPERATING CHARACTERISTICS OF THE OUTER SHIM CONTROL DRIVES

Length of Hafnium Poison Insert	46 inches
Control Cylinder Speed	0.055 rpm (20 degrees/minute)
Control Cylinder Travel	160 degrees (0 degrees full in 160 degrees full out)
Drive Motor	
Type	Instant reversing, three phase induction
Shaft Speed	2.2 rpm
Voltage	208 Volts ac
Seal	Rotary face type

1.2 Materials of Construction

Materials of construction for critical components of the outer shim control drive mechanism are shown in Table VIII-3-2.

TABLE VIII-3-2

CONSTRUCTION MATERIALS OF OUTER SHIM CONTROL DRIVE

Coupling Shaft	17-4 PH Stainless Steel
Coupling Shaft Spring	Inconel-X
Worm Shaft	Stellite 6B
Worm Gears	17-4 PH Stainless Steel
Worm Shaft Bearings	Stellite (ball)
Worm Gear Shaft Bearings	Stellite (ball)
Gear Box	Cast Aluminum
Drive-Shaft Housing Assembly	Type, 304 Stainless Steel
Drive-Shaft Bearing	Stellite 6

1.3 Access and Maintenance

The eight drive mechanisms penetrate the reactor vessel and biological shield radially to the outer shim control drive service area as shown in Figure VIII-1-1. The drive motors and outer gear box section can be serviced in this area during shutdown. Removal of the horizontal drive train is accomplished through this area.

The inner worm gear boxes and control cylinders can be removed through the reactor vessel.

1.4 Seal

The mechanical rotating face seal is flushed with a seal-water supply of demineralized water at higher than reactor pressure to prevent reactor coolant from contacting seal faces. A backup lip seal connected to a closed drain prevents reactor coolant from entering the service area in the event the face seal should fail.

1.5 Cooling

Reactor coolant flow holes are provided in the inner worm gear housing to flush and cool in internals. Cylinder cooling by reactor coolant is provided by the annulus between the cylinders and beryllium reflector and coolant passages in the cylinders.

1.6 Position Indication and Limit Control

The direct reading position transmitter provides a signal to the Decitrak system yielding a visual position indication (± 1 degree) of the cylinders at the console. Positive indication for the full-in position of individual cylinders is obtained hydraulically by a coolant porting system integral with the support beam and drive mechanism. The porting of this coolant line actuates a differential pressure switch, producing an electrical signal when the cylinder is fully inserted.

Mechanically actuated limit switches, double-pole, double-throw type, actuate signal lamps and relays cutting power to the motor. The switches are also used for interlocks on the control system.

Mechanical stops which positively limit cylinder over-travel are provided both in the individual external drive units and on the internal upper cylinder attachments. The slip clutch protects the mechanism from excessive torque.

The potentiometer signals are only used by the data processing system.

2. DESIGN CONSIDERATIONS

The primary design problems associated with the outer shim control drive mechanism were resolved by material changes as determined by the test program.

The design has incorporated two safety features to prevent the possibility of an accidental rapid insertion of reactivity should the cylinders become disengaged from the drive mechanism. These features are:

- (1) Failure of a connector column will permit the cylinder to drop slightly onto a bearing, the edge of which is sawtooth. The sawtooth edge bearing permits rotation only in the direction of decreasing reactivity.

(2) Water flow through the cylinder is designed to provide a torque for rotation in the direction of decreasing reactivity.

In addition to the above safety features, a hydraulic positioning indicator will provide a signal in the event the cylinder drops from normal position.

Additional design considerations to minimize possible mechanism malfunctions and ensure desired performance include:

Positive indexing of splined connections using missing tooth method

Total enclosure of springs

Moisture and overload protection for all electrical components

Drive motor high temperature capability (356°F)

Captivation of threaded connectors using class A locking cups.

3. MALFUNCTIONS

3.1 Electrical Power Failure

Failure of all the several power sources available to the drive systems is not considered credible because of individual drive wiring runs. Failure to individual units is possible.

(1) Drive Motor

No control cylinder motion is possible until power is restored.

(2) Limit Switches

Control cylinders can still be operated as long as power to the motor is not interrupted.

(3) Position indicator Transmitter

The digital position reading is lost.

(4) Potentiometer

The signal to the computer is lost.

3.2 Limit Switch Malfunction

Cylinders can be operated until mechanical stop has been contacted or motor reversed manually.

3.3 Failure of External or Internal Mechanical Stop in Conjunction with Limit Switch Failure

Either set of mechanical stops is adequate to prevent cylinder over travel.

3.4 Excessive Leakage of Shaft Seal

Excessive leakage can be detected on the gland system flowmeter. In addition, the leakage monitoring system will alarm.

3.5 Seizure of Shaft Seal

The nature of rotary face type seals makes shaft seizure a remote possibility. If this should occur, operation of the mechanism would not be possible until the seal is replaced. Removal and replacement of a seal is accomplished with a special nozzle sealing tool when the reactor is shut down and depressurized.

3.6 Seizure of Drive-Shaft Bearings

Seizure of either of the drive-shaft bearings would cause the mechanical clutch to slip preventing cylinder motion. Correction of a totally seized drive shaft is done in the same manner as described for replacing a seal. A partially seized shaft, caused by crud accumulation, could possibly be corrected by careful operation of the drive mechanism using an in-out method to free the shaft.

3.7 Seizure of the Worm Gears or Bearings

Seizure of any of the worm gear housing components results in the same situation as described above. Replacement is accomplished through the reactor head openings.

3.8 Failure of the Coupling Bolt

Although the coupling bolt is designed to withstand loads greater than the expected static and fatigue loads, the bolt may be broken by excessively large loads such as that which may be encountered during an attempt to remove a bolt with galled threads. Should total bolt failure occur, the control cylinder could drop until it is resting on the lower support bushing. A full-in indicator would signal indicating a malfunction. The cylinder would either remain in position or rotate in the direction of negative reactivity because of the saw edge on the bushing and the water torque through the cylinder. The control cylinder would have to be removed from the reactor.

3.9 Breakage of the Coupling-Shaft Spring

Breakage of the coupling-shaft spring would present the possibility of disengaging the shaft which in turn would prevent cylinder motion. This would be detected by the hydraulic full-in dp switch not activating after a shutdown.

4. PROTOTYPE OUTER SHIM CONTROL ROD DRIVE TESTS

A prototype ATR outer shim control rod drive mechanism was subjected to extensive testing under conditions simulating the ATR environment.

The tests included:

Life cycle test

Environment tests

Misalignment tests

Position accuracy tests.

4.1 Life Cycle Test

The life cycle test consisted of operating the mechanism 5100 cycles (full-in to full-out to full-in) under simulated ATR conditions. The coolant temperature was varied from about 90 to 180°F during the test period. In addition to the hydraulic and gravity cylinder loads, a radial load of 400 pounds was also applied to the cylinders during the test.

Results of this test showed satisfactory performance of the entire mechanism and individual components. The number of cycles and applied loads equals or exceeds expected reactor requirements.

4.2 Environment Tests

The drive mechanism was subjected to three environment tests of about 100 hours duration each. The mechanism was placed in a simulated ATR environment for a period of 100 hours without operation. The unit was then cycled 20 times and the performance observed. The test was repeated with the mechanism rest position changed.

Results showed satisfactory mechanism performance.

4.3 Misalignment Tests

Misalignment testing of the mechanism included horizontal and vertical drive misalignment, and combined misalignment testing. Imposed drive line misalignment was one inch vertically and one inch laterally. Cylinder misalignment was induced by a 0.007 inch/inch slope across the end of the cylinder in the area of contact with the column connector.

Results of these tests (5000 cycles) showed satisfactory performance of the mechanism under various conditions of misalignment.

4.4 Position Accuracy Tests

The backlash was measured before and after the above tests and is shown in Table VIII-3-3.

The hydraulic full-in indicator varied approximately one degree before and after the life cycle test.

TABLE VIII-3-3

MECHANISM BACKLASH
(Equivalent cylinder
rotation-degrees)

Test	Before	After
Life Cycle (5110)	0.4 to 0.5	0.9
Environmental (300 hours-60 cycles)	0.9	0.9
Misalignment + Loads + Temperature	0.0	0.9

5. SUMMARY

The ATR outer shim control rod drive mechanism has performed satisfactorily under extensive testing in an environment simulating ATR conditions. Misalignment tests have shown the mechanism capable of satisfactory performance under conditions more severe than expected in the reactor. All production components were inspected and found to be free of corrosion and wear.

Based on design considerations, performance testing, and malfunction analysis, the ATR outer shim control drive will meet reactivity control requirements in a reliable manner.

IX. ELECTRICAL SYSTEMS

- IX-1 Power System Arrangement
- IX-2 Power System Protection
- IX-3 Alarms and Communications

**THIS PAGE
WAS INTENTIONALLY
LEFT BLANK**

IX-1. POWER SYSTEM ARRANGEMENT

1. ELECTRICAL DISTRIBUTION FACILITIES

1.1 General Description

The electrical distribution facilities for ATR consist of five (5) separate systems as follows:

- (1) Commercial power
- (2) Diesel power
- (3) Diesel-commercial power
- (4) Instrumentation ac battery-backed power
- (5) Utility dc battery power.

The ATR and ETR reactors have a common source of commercial power from two existing 20,000-kVA transformers. Each transformer is energized by a connection to the 132-kV NRTS loop circuit. Both facilities are served by individual feeders from each transformer.

Because the commercial power system is subject to occasional outages, two 1500-kW diesel engine-generators are used as a second independent source of ac power. One unit operates continuously supplying power to reactor and experiment equipment essential for safe reactor shutdown due to loss of commercial power.

Diesel-commercial power is controlled from a motor control center which may be energized with either diesel or commercial power. Upon loss of either power source, the load is automatically transferred to the other. The diesel system is the normal power source.

Instrumentation ac battery-backed power is supplied by a dc-ac motor-generator set energized from a dc system consisting of an ac-dc motor-generator set energized with diesel power, and a floating battery. The battery supplies power upon loss of diesel source.

Utility dc power is supplied by an ac-dc motor-generator set energized with diesel power. A floating battery supplies utility dc power upon loss of diesel source.

Power is distributed at the voltages outlined in Table IX-1-1.

The electrical distribution systems are shown on the following one-line wiring diagrams:

Figure IX-1-1 Main One-Line Wiring Diagram

Figure IX-1-2 Auxiliary One-Line Wiring Diagram, Sheet No. 1

TABLE IX-1-1

ELECTRICAL DISTRIBUTION VOLTAGES

Voltage	Comm.	Diesel	Diesel-Comm.	Battery Backed	
				ac	dc
13,800 ac, 3 phase	x				
4,160 ac, 3 phase	x	x			
480 ac, 3 phase	x	x	x		
120/208 ac, 1 & 3 phase	x		x	x	
250 ac					x

Figure IX-1-3 Auxiliary One-Line Wiring Diagram, Sheet No. 2

Figure IX-1-4 250-V DC and 120/280-V AC One-Line Wiring Diagram

Figure IX-1-5 Auxiliary One-Line Wiring Diagram-Connections to Existing Facilities

1.2 Commercial Power

The commercial power system consists of the following components:

1.21 13,800 Volt. ATR 13,800-volt feeders originate at the two ETR transformers sharing terminals with existing ETR feeders. Metal-clad, 500-mVA switchgear is installed in Control House MTR 623 and consists of a breaker for each feeder and a tie breaker between the feeders. Remote breaker control and metering of electrical power are provided at the NRTS Central Facilities control board.

Each of the two feeders supplies a separate ATR 7500-kVA transformer located outside the reactor building. These transformers furnish power at the 4160-volt level to two sections of distribution switchgear in the building.

During an outage involving one main ETR transformer, the corresponding ATR 13,800-volt feeder may be energized from the remaining transformer through the normally closed ATR tie breaker.

1.22 4160 Volt. The two sections of 4160-volt, 150-mVA switchgear control motors rated 150 hp and above and feeders to transformers supplying power to 480-volt switchgear.

The sections of 4160-volt switchgear have a normally open tie switch. This tie switch permits energizing both sections for limited loading upon failure of either transformer circuit. Because the switch must not be operated under load, its enclosure has a special lock, the key being available only to authorized

personnel. Before energizing 4160-volt switchgear, through the tie switch, the faulty transformer must be disconnected by removing a section of the 4160-volt open bus.

1.23 480 Volt. The two sections of 480-volt switchgear in the reactor building control motors rated 60 to 125 hp, feeders to 480-volt motor-control centers which control reactor and building operating equipment, and 480-volt power centers which control feeders to experiment motor-control centers.

The 480-volt switchgear in the ATR cooling tower pump house controls the cooling tower fan motors and the feeder to a 480-volt motor-control center serving the cooling tower area.

The 480-volt motor-control centers consist of standard combination air circuit breaker and electrically operated magnetic contactor starters for motors rated 1/2 to 50 hp, and individual air circuit breakers for feeders to power and lighting loads.

The 480-volt power centers consist of air circuit breakers for feeders to experiment motor-control centers.

1.24 120/208 Volt. Lighting panels are supplied through 480-120/208-volt transformers energized from the 480-volt motor-control centers. They supply power for lighting, motors rated below 1/2 hp, and other small loads.

1.3 Diesel Power.

Diesel power at 4160 volts ac is normally supplied by one of the two 1500-kW diesel-engine generators, either of which can carry the full diesel load. The alternate unit can be started and synchronized to the bus for the purpose of load transfer between units.

1.31 4160 Volt. The 4160-volt, 75-mVA switchgear controls the two diesel generators, motors rated 150 hp and above, and the feeder to a 480-volt unit substation. The diesel engines are controlled from engine start-up instrument panels located adjacent to the diesel generator units. The generators are controlled normally at the switchgear and, in emergency, from a panel located near the diesel engine start-up instrument panels.

The switchgear has a spare circuit breaker position into which a breaker may be inserted. The closing of this breaker ties the diesel bus to a section of 4160-volt commercial bus, which permits energizing selected commercial loads upon failure of the normal source. To maintain the integrity of the separate electrical systems during reactor operation, this compartment has a special lock, the key being available only to authorized personnel.

1.32 480 Volt. The 480-volt switchgear controls motors rated 60 to 125 hp, feeders to 480-volt motor-control centers for reactor and building operating equipment, and 480-volt power panels controlling feeders to experiment motor-control centers.

The 480-volt motor-control centers consist of standard combination air circuit breaker and electrically operated magnetic contactor starters for motors rated 1/2 to 50 hp, and individual air circuit breakers for feeders to power

loads. The 480-volt power panels consist of air circuit breakers for feeders to experiment motor-control centers.

1.4 Diesel-Commercial Power

Diesel-commercial power at 480 volts ac is normally supplied from the 480-volt diesel power system.

Upon loss of diesel power, load is automatically transferred by electrically interlocked circuit breakers to the 480-volt commercial power system. The transfer is rapid enough so that the service is essentially continuous. When diesel power is restored, the system is automatically transferred back to diesel power.

1.41 480 Volt. A 480-volt, diesel-commercial, motor-control center controls power for essential pumps, reactor control, and loop instrumentation loads which require their supply to be automatically maintained from either commercial or diesel power.

1.42 120/208 Volt. A 120/208-volt, diesel-commercial distribution panel, energized through a transformer from the above motor-control center, supplies power for diesel-commercial lighting, reactor and process control panels, data processing, paging, reactor warning alarm systems, and health physics monitoring.

1.5 Instrumentation AC Battery-Backed Power

Instrumentation ac battery-backed power at 250 volts is normally supplied by a dc-ac motor-generator set energized from a dc system consisting of an ac-dc motor-generator set, energized with diesel power, and a floating battery. In the event of loss of the diesel power driving the ac-dc motor-generator set, the battery will maintain the operation of the dc-ac motor-generator set for an emergency period of a minimum of 1/2 hour.

A standby transformer energized from commercial power can be manually switched to supply unregulated ac power for instrument operation during reactor shutdown. This transfer can be made only if the dc-ac motor-generator set generator breaker is open.

The instrumentation ac battery-backed power-control center supplies power at 120/208 volts ac for the reactor control and protection instrumentation and for the data processing system electronics. Local instrument rectifiers provide dc required for reactor instrumentation.

1.6 Utility DC Power

Utility dc power at 250 volts is normally supplied by an ac-dc motor-generator set energized with diesel power and backed up by a floating battery. The battery will maintain the dc power loads (the primary emergency coolant pump) in operation for a minimum of 1/2 hour upon loss of the normal motor-generator source. The utility dc power-control center controls the dc primary emergency coolant pump motor and feeders for switchgear operation and control. The battery is independent of the battery used for supply of ac instrumentation. The actual life of battery operation will be field tested prior to operation.

1.7 Existing Power System Extensions

A new circuit breaker has been added to the existing Steam Plant 2400-volt, normal-emergency bus switchgear for a feeder to the existing ETR plant air compressor.

A new feeder has been added to an existing circuit from the Raw-Water Pump House existing 2400-volt normal bus for two new 480-volt motor-control centers to serve Raw-Water Pump House and Well Pump House loads.

New circuit breakers have been added to Demineralizer Building, 480-volt motor-control center and 120-volt lighting panel to serve new demineralizing equipment loads.

All of these circuits are controlled and protected in a manner similar to that for ATR.

1.8 Emergency Lighting

Lighting units of alternate areas of the reactor building are energized with commercial and diesel power, respectively. All units are normally energized. On loss of either power source, the units of the alternate power source maintain a degree of illumination. Should both power sources fail, battery-operated lights at selected locations provide safe egress illumination.

The battery-operated light units each consist of two lamps energized from a storage battery. The batteries are kept charged by continuous float charging from the diesel-commercial power source. The lamps are automatically switched on upon failure of both sources of ac power.

FIG. IX-1-1 MAIN ONE-LINE WIRING DIAGRAM.

<u>REFERENCE DRAWINGS:</u>	ATR-1075-MTR
AUXILIARY ONE LINE WIRING DIAGRAM	
SH. 1	670-E-21
SH. 2	670-E-22
MAIN ONE LINE WIRING DIAGRAM	406

FIG. IX-1-4 250-V DC AND 120/208-V AC ONE-LINE WIRING DIAGRAM.

IX-2. POWER SYSTEM PROTECTION

Electrical system components are subject to damage from lightning, overheating of current carrying parts, and physical breakage, all of which are potential causes of loss of power. Overheating of current carrying parts may occur inadvertently by overloading or accidentally by failure of insulation, the latter being referred to as a fault. The accidents considered, together with preventative measure, are given below.

1. LIGHTNING PROTECTION

The overhead portions of the two 13,800-volt ATR feeders are exposed to direct lightning stroke. To protect the lines against lightning damage, an effectively grounded shield wire is provided over the lines. To protect the terminal equipment against high-voltage surges, lightning arresters are provided at each end of the lines and at both high- and low-voltage bushings of the ATR outdoor transformers.

2. OVERLOAD PROTECTION

2.1 ETR Transformers

The ETR transformers are protected by overcurrent relays against external overloads and faults. These relays operate on an inverse time basis, ie, the heavier the overload, the faster the relays operate. In addition, differential relays are provided to isolate a transformer in case it develops internal faults and reverse relays to prevent current flow from a sound 132-kV section to a faulted 132-kV section.

The relays operate to open the associated 13,800-volt ATR circuit breakers and are an extension of the existing ETR protective scheme to cover the ATR facilities. The opening of an ATR 13,800-volt circuit breaker is annunciated at the NRTS Central Facilities control board.

A tie circuit breaker is provided between the two 13,800-volt ATR feeders which permits carrying the ATR facility on one ETR transformer should the parallel transformer be out of service.

2.2 Conductors

Normal load currents are based on the forced-air-cooled rating of transformers and on standard ratings of motors including service factors when applicable.

Conductor sizes have been selected to prevent fusing by fault currents before the relays act to clear faulted sections.

Rigid bus conductors are suitably supported to prevent distortion resulting from high fault currents.

2.3 7500-kVA ATR Transformers

These transformers are protected by overcurrent relays and differential relays in a manner similar to that described for the ETR transformers. These relays trip the 13,800-volt circuit breakers controlling the transformer circuit in trouble. Additional protection is obtained by annunciation at the process control board if either transformer has high oil temperature, low oil level, expansion tank high pressure or high vacuum.

2.4 Switching Equipment

All switching equipment, including switchgear, motor-control centers, and power panels, are provided with protective devices. These devices, consisting of relays or thermal overload elements, serve to detect overload or fault conditions on individual feeders and isolate the faulted feeder, thereby maintaining continuity of service for the remaining electrical systems.

2.5 Motors

A circuit breaker with instantaneous and time delay tripping is provided in each motor feeder close to the source of power. The instantaneous element serves to isolate the feeder in case of a fault and to protect the feeder conductor and motor against high fault currents. The time delay elements, which are adjusted carefully to match the thermal characteristics of the motor, protect it against excessive heating caused by overloading.

2.6 Diesel Generators

The diesel generators are protected against overloads and internal faults by differential relaying and ground fault relaying.

3. SERVICE RELIABILITY

The following features prevent damage to electrical components and subsequent damage to the reactor:

(1) Loss of Power

Loss of commercial power at the 4160-volt level initiates a reactor scram. The loss will be detected by under voltage relays on both sections of 4160-volt switchgear bus. The relays will ignore harmless voltage fluctuations but will react quickly to abnormally low voltage to signal loss of power. The voltage at which the relay operates is adjustable. The scram signal on loss of power can be transmitted in less than 40 milliseconds.

(2) Power Distribution

The commercial power system is so arranged that at 4160 volts and below, power is distributed through two effectively separate systems. Relaying is provided to isolate one system in case of failure and thereby prevent loss of power in the other system. All main reactor operating equipment motors energized with commercial power, if more than one unit is involved, are divided equally between the two systems. This principle applies to primary and secondary coolant pumps, primary pressurizing pumps, and similar services.

(3) Multiple Power Source Availability

Motors for the following essential equipment may be energized with either commercial power or diesel power, as manually selected for each motor:

- 2 - Primary pressurizing pumps
- 2 - HDW circulating pumps
- 2 - Primary gland seal pumps
- 3 - NT pressurized water loop primary circulating pumps
- 15 - Pressurized water loop primary circulating pumps

(4) Codes

To meet the requirements of reliability of service and safety in operation, commercially available products made under applicable ASA, NEMA, and other codes and standards are used.

(5) Fire Protection

To confine potential fire hazards, cable is flame retardant and is installed in isolated groups. Switchgear and motor control centers are metal-enclosed with separating metal barriers between units.

(6) Protective Relays

Protective relays, in conjunction with fully rated circuit breakers, operate to isolate equipment or cable that is overloaded or is under fault condition, so that service remains intact to the remaining electrical equipment.

(7) Outside Service

All outdoor overhead installations are designed for heavy loading conditions of wind and ice as defined in National Electrical Safety Code.

(8) Underground Service

Underground ducts are encased in concrete and, for identification, are colored red.

(9) Manual Reset

Motor circuits open automatically on loss of power, and the breakers must be reset manually so that heavy loads may be restored in an orderly manner.

(10) Remote Station Lockout

Lockout pushbutton stations are provided for each motor that is remotely controlled to permit a maintenance man to interrupt the power circuit.

(11) Ventilation

A fire protection device stops the ventilation intake fan motor and closes the louvre damper to prevent oily smoke from being carried into the building from an outdoor transformer fire.

4. RESTORATION OF ESSENTIAL SERVICES AFTER BUILDING EVACUATION

A forced evacuation of the reactor building may be accompanied by loss of electrical service. In this event, an emergency electrical-diesel generator start-up control panel (ESP) permits energizing the diesel bus from one diesel generator and the restoration of essential services.

4.1 General Description

The ESP panel is located outside the gas-tight area at the foot of the stairs leading to the diesel generator area. This location is near the engine start-up instrument panels. The area is partially shielded by walls and floor against radiation from the gas-tight area. The panel is located and arranged so that necessary operations can be performed in a minimum of time. The ESP panel controls also override all radiation interlocks and synchronizing procedures. Therefore, it is enclosed in a metal cabinet which is provided with a key locked door. The key to the door lock is available by breaking a small glass panel in the enclosure.

The ESP panel is provided with switches for the control of generator and essential load circuit breakers, with red and green breaker-position indicating lights and with white potential indicating lights.

Connecting a generator to the 4160-volt bus will automatically reenergize the diesel system since feeder breakers do not open on loss of power. Light load essential services will automatically be restored. Heavy essential services may be restored by control switches on the panel.

The operator will first start a diesel engine from the engine start-up instrument panel. The engine system is designed for starting with a hand priming fuel oil pump. Lube oil is circulated by the shaft driven pump. Jacket water will be available from the fire-water backup. Starting air for at least eight cold starts is available.

After a diesel generator is started, the 4160-volt diesel bus is energized by closing the generator field breaker and generator main breaker by means of controls at the ESP panel.

4.2 Equipment Automatically Restored and Manually Started

When the bus is energized, the following services will automatically be restored:

Warm Seal Water Booster Pumps 670-M-82 & 83

Primary Gland Seal Pumps 670-M-40 & 41

Diesel Engine Starting Air Compressor 670-M-65

Diesel Engine Auxiliary Jacket Water Pumps 670-M-60 & 61

Diesel Engine Standby Lubricating Oil Pumps 670-M-48 & 49

Warm Waste Sump Pump 670-M-31

AC Primary Emergency Coolant Pump 670-M-10

Diesel-Commercial Lighting

Process Control Panel Power

Reactor Control Panel Diesel-Commercial Power

Health Physics Monitoring Panel Power

Communications and Alarms Power

The following services may be restored from the emergency panel by the operator:

Utility cooling-water pumps 671-M-6 & 5

Instrument dc power M-G set 670-E-25

Utility dc power M-G set 670-E-22

Primary pump motor area transfer fan 670-HVE-3B

Utility area supply fan 670-HVS-5

Main exhaust fan 670-HVE-17B

Reactor area supply fan 670-HVS-1

Reactor Operating area exhaust fan 670-HVE-18

The following items of equipment start automatically when the equipment, under which they are listed, is started.

Main exhaust fan 670-HVE-17B

Heat exchanger area exhaust fan 670-HVE-4 (if radiation monitors interlock permits)

Subpile & rod access rooms & waste tank area exhaust fan 670-HVE-8

Degassing tank exhaust fan 670-HVE-18

Reactor room exhaust fan 670-HVE-1B

Reactor annulus exhaust fan 670-HVE-7

Reactor Area Supply Fan 670-HVS-1

Reactor Operating Area Air Washer Pump 670-HVW-1

Reactor Operating Area Exhaust Roof Ventilator 670-HVR-1

First Basement Supplementary Transfer Fan 670-HVE-2

1st & 2nd Basement Cubicle Exhaust Fan -- West Side 670-HVE-5

1st & 2nd Basement Cubicle Exhaust Fan -- East Side 670-HVE-6

Piping Corridor Exhaust Fan 670-HVE-16

Utility Area Supply Fan 670-HVS-5

Utility Area Air Washer Pump 670-HVW-2

Utility Area Roof Ventilators 670-HVR-5, 6, & 7

Utility Area Office, Toilet & Water Lab Exhaust Fan 670-HVE-13

Utility Area Return Air Exhaust Fan 670-HVE-10

Battery Room Exhaust Fan 670-HVE-9

In case of loss of commercial power without any ATR fault condition, commercial power may be restored by the NRTS Central Facilities operator by reclosing 13,800-volt feeder breakers. This automatically restores commercial power at all voltage levels, and permits normal operational control at the process control panel.

IX-3. ALARMS AND COMMUNICATION

1. ALARM SYSTEMS

1.1 Fire Alarm

The ATR fire alarm system is an extension of the coded ADT system now in use at ETR and MTR which sounds alarms at NRTS Central Facilities as well as locally.

1.2 Evacuation Alarms

Evacuation alarms are manually originated by switches in the reactor control room and sound alarms for evacuation of (a) ATR area and (b) MTR-ETR area. Both switches have "Alert" and "Evacuate" positions with a spring return to midposition. "Alert" position sounds the sirens continuously. "Evacuate" sounds the sirens intermittently. Sirens at ATR can also be sounded from switches in the MTR-ETR area.

A special direct telephone line connecting the ATR supervisor with ETR and MTR supervisors can also be used to communicate and direct evacuation procedures.

1.3 Reactor Alarms

Signals are originated from the reactor control panel to indicate "Reactor On". A continuous visual indication is given on the reactor control panel, the process control panel, the control panel in Demineralizer Building 609 extension and other operating locations in the ATR building.

1.4 Other Alarms

Alarms associated with operation of the reactor and process system and the radiation monitoring system are described in Section VII.

2. COMMUNICATION SYSTEMS

2.1 Telephone

The ATR system is an extension of the existing Mountain States telephone system in the ETR and MTR areas. This permits dialing from one extension to another and audible code-calling by dialing a code number.

2.2 Paging

A paging system within the ATR facilities provides for audible paging, public address, and telephone conversation.

2.3 DC Powered Telephone

A system within the ATR facilities, using dc powered carbon-type telephones, permits communication between two or more preconnected stations.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

X. ADDITIONAL COOLANT SYSTEMS

- X-1 Secondary Coolant System**
- X-2 Utility Cooling-Water System (UCW)**
- X-3 High-Pressure Demineralized Water System (HDW)**
- X-4 Secondary Coolant System Instrumentation**

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

X-I. SECONDARY COOLANT SYSTEM

1. GENERAL

The secondary coolant system is a circulating treated-water system which removes heat from the reactor primary coolant in the primary heat exchangers and rejects the heat to the atmosphere through a cooling tower.

The secondary coolant system includes the shell side of the primary heat exchangers, an induced draft cooling tower, four vertical, deepwell, turbine-type secondary coolant pumps, cooling tower chemical treating equipment, and associated carbon steel piping and valves. A flow diagram of this system is shown in Figure X-1-1.

The four 8600-gpm secondary coolant pumps discharge into a common 36-inch header to provide a total capacity of 34,400 gpm. They are mounted vertically in individual bays of the coldwell of the cooling tower basin with a minimum submergence of 26 inches. An extra bay is included for a future spare pump. The four pumps, during summer operating conditions with the reactor at full power [250 MW(t)], supply the required 31,200 gpm of 85°F cooling water. Secondary water is pumped through a butterfly-type control valve to the shell side of the primary heat exchangers, in which the secondary coolant-water temperature rises to 138°F and is returned to the cooling tower. A mechanical stop is installed on the control valve to maintain at least 2000 gpm in the minimum flow position to ensure flow to the primary heat exchangers at all times and to prevent overheating of the secondary coolant pumps.

Provisions are made for emergency circulation of water in the secondary coolant loop by use of utility cooling-water pumps. These pumps also utilize the cooling tower as a heat sink and provide protection in the event that the secondary coolant pumps are disabled by loss of commercial power, since one pump is operated continuously and the other automatically started on low secondary coolant pressure and is powered from the diesel bus. The standby utility cooling-water pump will start automatically on low utility water discharge header pressure. A pipe interconnection is provided between the utility coolant pump discharge header and the secondary cooling-water pump discharge header for this purpose. A control valve in this interconnection opens automatically on loss of commercial power. Commercial power loss causes a reactor scram. This valve, capable of passing 2000 gpm, is provided with a minimum adjustable stop initially set at 1500 gpm. This flow is adequate to remove reactor decay heat after shutdown.

The secondary coolant flow control valve is positioned by primary temperature control signals, described in Section VII, to maintain the reactor primary coolant-inlet temperature within $\pm 1.5^\circ\text{F}$ of its setpoint in the range 125 to 130°F.

2. WATER TREATMENT

The secondary coolant water is treated with sulphuric acid to maintain a pH of 6.3, with chlorine solution to minimize biological fouling, and with phosphate solution to inhibit carbon steel corrosion. Cooling tower makeup and blowdown are controlled to limit the total dissolved solids in the secondary coolant to 1000 ppm, or about four concentrations of the well water composition. This combination of treatments produces a water composition which limits the rate of deterioration of both the carbon steel equipment and the cooling tower frame and fill.

3. DETECTION OF PRIMARY HEAT-EXCHANGER LEAKS

A radiation monitor is installed in each secondary coolant-outlet line from the primary heat exchangers to detect the leakage of radioactive primary coolant into the secondary coolant system.

4. COOLING TOWER MAKEUP

Two 2500-gpm cooling tower makeup pumps, located in an extension of the existing MTR raw-water pump house, take suction from the existing ground level water-storage tanks and discharge to the cooling tower basin. In addition, 600 gpm of well water, available for use by the cubicle air coolers, is discharged to the cooling tower basin for makeup. The cooling tower makeup instrumentation is described in Section VII.

5. EQUIPMENT DESIGN

5.1 General

The secondary system is designed, fabricated, and erected in accordance with ASME and ASA codes and standards. The work was inspected by qualified inspectors with experience in the types of equipment and construction procedures used. The equipment specified is high quality, conventional commercial products of proved reliability and incorporates materials of construction known to give good service under the proposed conditions of operation.

5.2 Equipment Performance

The design conditions of the major secondary coolant system equipment are as follows:

(1) Cooling Tower

Design Characteristics

Water Flow

38,200 gpm

Wet bulb temperature	65°F
Water exit temperature approach to wet bulb temperature	20°F
Water entering temperature	136.5°F
Water exit temperature	85°F
Cooling range of water	51.5°F
Cooling duty	9.76×10^8 Btu/hr

Materials of Construction

Tower frame and fill	Treated redwood
Anchor bolts	Stainless steel
Joints connectors	Naval bronze, glass reinforced polyester resin, or galvanized steel
Nails	Bronze and galvanized
Fan blades	Forged aluminum
Spray nozzles and splash plates	Plastic

Auxiliary Features

The cooling tower is equipped with a deluge system for fire protection which is supplied from the ATR fire-water system.

(2) Secondary Coolant Pumps (Four Units in Parallel)

Design Characteristics

Type	Vortical deepwell turbine
Capacity, each	8600 gpm
TDH	145 feet
Minimum submergence	26 inches
Suction temperature	85°F
Speed	1200 rpm
Brake horsepower	375 hp
Motor horsepower	350 hp

Materials of Construction

Discharge head	Cast iron
Impellers	Bronze
Bowls	Cast iron
Casing	Cast iron
Shaft	SAE 1045 steel

X-2. UTILITY COOLING-WATER SYSTEM (UCW)

1. PURPOSE AND ARRANGEMENT

The utility cooling-water system utilizes water from the cooling tower to absorb heat from the experimental loops via the high-pressure demineralized water system (HDW) and from the diesel generator auxiliaries. In addition, the utility cooling-water system is arranged to remove reactor-decay heat in the event the secondary coolant flow is lost. A flow diagram of this system is shown in Figure X-2-1.

2. DESIGN FEATURES

Two 3050-gpm (one a spare) vertical, deepwell, turbine-type utility cooling-water (UCW) pumps are mounted in separate bays of the coldwell of the cooling tower basin. The pumps discharge cooling water into a common header, thence through the heat exchangers, and returning to the cooling tower. Both UCW pumps are supplied from the diesel power bus. In addition to other duties, each UCW pump has sufficient capacity to supply cooling water to both diesel engines so that both engines can be operated simultaneously at any time.

After a reactor scram coupled with the loss of secondary coolant pumps, the UCW flow to the HDW system is reduced to 1050 gpm to ensure a supply of 1500 gpm to the primary heat exchangers.

The UCW system is similarly backed up in the event of loss of diesel power by the secondary cooling-water system. A check valve in an interconnection between these systems is normally held closed by the higher operating pressure in the UCW discharge header, but opens automatically if the UCW water pumps fail to maintain their normal discharge pressure. On loss of diesel power, a reactor scram occurs; hence, the secondary coolant pump capacities are in excess of that needed for reactor decay heat removal and the UCW system.

3. SPECIAL FEATURES

During low power flux mapping operations, UCW may be utilized instead of secondary coolant for secondary heat removal, since this enables more accurate control of primary coolant temperature. To achieve this, the control valve in the interconnection between the systems is provided with a manual flow control loader on the Process Control panel.

In addition to the above emergency provisions, fire water is automatically admitted to the utility cooling-water line serving the diesel generator auxiliaries in the event of loss of all diesel power, which stops the UCW supply. This assures 500 gpm of cooling water to the diesel engine heat exchangers to permit the restarting of one diesel engine.

UCW FLOW BALANCE

NORMAL CONDITIONS

UCW TO ONE DIESEL ————— 500 GPM
 UCW TO HDW HEAT EXCHANGERS ——— 2050 GPM
 WITH TWO DIESELS RUNNING ————— TOTAL 3050 GPM

COMMERCIAL POWER FAILURE

UCW TO PRIMARY HEAT EXCHANGERS ——— 1500 GPM
 UCW TO ONE DIESEL ————— 500 GPM
 UCW TO HDW HEAT EXCHANGERS ——— 1050 GPM
 TOTAL 3050 GPM

DIESEL POWER FAILURE

SECONDARY TO HDW HEAT EXCHANGERS — AS MUCH AS DESIRED

ABBREVIATIONS
 CONT. - CONTINUATION
 EXCH. - EXCHANGER
 CONN. - CONNECTION

FIG. X-2-1 UTILITY COOLING-WATER SYSTEM FLOW SHEET,

X-3. HIGH PRESSURE DEMINERALIZED WATER SYSTEM (HDW)

1. PURPOSE AND ARRANGEMENT

The high-pressure demineralized water (HDW) system cools the in-pile experiment loops by absorbing heat from the experiment primary heat exchangers, the experiment primary circulation pumping units, and the experiment purification systems. A flow diagram of this system is shown in Figure X-3-1.

2. DESIGN FEATURES

The HDW system is a closed loop circulating treated demineralized water at 200 psig. The system provides 130°F water to the shell side of the experiment heat exchangers, primary circulating pumping units and experiment purification systems. Heat absorbed by the HDW system is rejected in shell and tube heat exchangers to the UCW system.

FIG. X-3-1 HIGH PRESSURE DEMINERALIZED WATER SYSTEM FLOW SHEET.

X-4. SECONDARY COOLANT SYSTEM INSTRUMENTATION

1. GENERAL

The secondary coolant system instrumentation records, indicates, controls, and provides off-normal alarm signals for the following:

Secondary Coolant System

Utility Cooling-Water System

High-Pressure Demineralized Water System.

2. COOLING TOWER MAKEUP CONTROL (See Figure X-1-1)

Normal level in the cooling tower basin is maintained by makeup water supplied by the cooling tower makeup pumps described in this section. Normally only one pump is operated, the second pump being on standby. Either pump can be started or stopped manually from the 480-V Normal Motor Control Center 619-E-2, located in the raw-water pump house.

A level control valve, LCV-2-1, located in the cooling tower makeup line is actuated by the indicating basin-level controller, LIC-2-1, mounted on the process control panel. This control valve closes upon loss of instrument air supply pressure to avoid flooding the cooling tower basin and surrounding area.

High- and low-level alarm annunciator (LA-2-1) and makeup water-flow recorder and totalizer FR-2-3 and FQ-2-3 are on the process control panel.

3. SECONDARY COOLANT BLOWDOWN CONTROL (See Figure X-1-1)

A blowdown system is provided to maintain the suspended and dissolved solids in the secondary coolant system at 1000 ppm. The blowdown flow rate is controlled by flow control valve FCV-2-2 which is actuated by manual loader FRML-2-2 located on the process control panel. This control valve closes upon loss of instrument air-supply pressure to maintain maximum water level in the cooling tower basin since makeup control valve LIC-2-1 also closes on loss of instrument air-supply pressure. A blowdown flow totalizer, FQ-2-2, is mounted on the process control panel.

4. PRIMARY HEAT EXCHANGER SECONDARY SIDE TEMPERATURE (See Figure X-1-1)

Four resistance temperature bulbs (three average, one spare) are installed in each outlet of the shell side of the primary coolant heat exchangers. Similarly,

four bulbs are provided on the secondary coolant inlet and outlet lines. All six points are recorded every 24 seconds on TR-2-1 located on the process control panel. This recorder has an accuracy of $\pm 1/4$ percent and fails upscale on element or amplifier failure. Differences in the outlet temperature of one of the exchangers indicate possible tube plugging or surface fouling.

5. SECONDARY COOLANT-WATER POWER MEASUREMENT

The secondary coolant-water power measurement is obtained by multiplying the linear flow signal by the secondary coolant temperature rise. The secondary coolant-flow recorder, FR-2-1, temperature-rise recorder, DTR-2-4, and the water-power recorder controller, WRC-2-1, are mounted on the process control panel.

6. UTILITY COOLING-WATER SYSTEM INSTRUMENTATION

(See Figures X-1-1 and X-2-1)

Two utility cooling-water pumps provide cooling water to various system heat exchangers. The standby UCW pump automatically picks up the load upon failure of the first pump. Either pump can be started or stopped manually from the process control panel. Automatic starting is accomplished by two pressure switches, PS-2-14A and PS-2-14B, on the pump discharge header. The individual pump discharge pressures are indicated by locally mounted pressure gauges PI-2-5 and PI-2-6. Low-pressure alarm PA-2-15 on the pumps discharge header is annunciated on the process control panel.

Flow control valve FCV-2-4 is installed in the connecting piping between the utility cooling-water header and the secondary cooling-water inlet header to the primary heat exchangers. This valve is closed during normal operation. In case of loss of commercial power, control valve solenoid FJV-2-4 de-energizes and the flow control valve FCV-2-4 opens, permitting the UCW system to supply the cooling-water flow necessary for removing decay heat from the primary system. Flow control valve FCV-2-4 opens on loss of instrument air to ensure cooling water for removing reactor decay heat. Also, on loss of commercial power the UCW flow to the HDW heat exchangers is limited to 1050 gpm in order to assure UCW being available to remove the decay heat from the primary coolant system.

During flux mapping and low-power operation, recording-manual loader FRML-2-4, located on the process panel, is used to maintain primary coolant temperature by controlling the flow rate through FCV-2-4.

On loss of both commercial and diesel power, the UCW supply header is interconnected automatically with the fire-water supply to provide fire-water cooling for the diesel generator exchangers. Pneumatic control valve PCV-4-7 is installed in the interconnecting piping between the Fire Water Main and the UCW line to the diesel generator. The valve is controlled by pressure switch PS-4-7 which senses the loss of utility cooling-water header pressure. The

control valve is closed by air pressure and is spring opened. It is closed during normal operation, but opens on loss of instrument air.

When the utility cooling-water pressure is restored to normal by either the UCW or secondary coolant pumps, the solenoid valve will be energized; however, control valve PCV-4-7 will not close unless the operator manually resets solenoid PJV-4-7.

The fire-water valve position switch annunciates alarm PVA-4-7 on the process panel to advise the operator that the control valve is open.

7. HIGH PRESSURE DEMINERALIZED WATER (HDW) SYSTEM INSTRUMENTATION (See Figure X-3-1)

The purpose of this system is to provide treated demineralized cooling water to the experiment equipment. The system consists of one HDW expansion tank and two HDW circulation pumps.

7.1 HDW Expansion Tank

The HDW circulation pump suction is pressurized to 200 psig by the HDW expansion tank. The tank is pressurized manually using compressed air from the 250-psi hp air system. This tank dampens pressure surges by compression and expansion of the air cushion.

The tank pressure is indicated on pressure indicator PI-7-2 on the process control panel. The tank high or low pressure alarm PA-7-2 is annunciated on the process control panel.

The indicating level controller, LIC-7-2, maintains the HDW system inventory by controlling the makeup control valve, LCV-7-2A, and the system bleed control valve, LCV-7-2B, using split range positioners. The makeup control valve, LCV-7-2A, opens on air failure, and bleed control valve, LCV-7-2B, closes on air failure to ensure an adequate water supply to the HDW circulation pumps.

The tank high or low level alarm LA-7-2 is annunciated on the process control panel.

7.2 HDW Circulating Pumps

Under normal operation only one pump is running. The second pump is a standby which starts automatically on low discharge pressure actuated by pressure switches PS-7-6A and PS-7-6B on the pump discharge header.

The pump discharge pressure is indicated by PI-7-5 on the process control panel. The pump high or low discharge alarm PA-7-5 is annunciated on the process control panel.

7.3 HDW Heat Exchangers

HDW is supplied to the loops at two temperatures, 130 and 100°F. The flow of 130°F HDW is indicated on flow indicator FI-7-2 on the process control panel. The flow of 100°F HDW is indicated on flow indicator FI-7-3 on the process control panel.

In order to maintain constant HDW pressure to all the experiments, control valve PCV-7-1 is installed in the connecting piping between the pump discharge and the pump suction header, which is pneumatically actuated by the locally mounted indicating pressure controller PIC-7-1. In order to maintain flow to the experimental equipment, this valve fails closed on loss of instrument air. The constant HDW pressure permits readjustment of one HDW coolant branch without affecting the others.

A locally mounted temperature indicating switch, TIS-7-7, measures the outlet temperature from the 130°F HDW heat exchangers. A high temperature alarm, TA-7-7, is also annunciated on the process control panel. A locally mounted temperature indicating switch, TIS-7-8, measures the outlet temperature from the 100°F HDW heat exchanger. High temperature alarm TA-7-8 is also annunciated on the process control panel.

The flow of UCW to the HDW heat exchangers is 2050 gpm during normal operation and is indicated on FI-4-2 located on the process control panel.

On loss of commercial power this flow is reduced to 1050 gpm by control valve FCV-4-5. During normal operation this control valve is wide open, but on loss of commercial power, solenoid valve FJV-4-5 permits the valve to close to a preset stop which limits flow to 1050 gpm.

The HDW heat exchanger temperature is regulated by a bypass control system. UCW flow around the heat exchangers is regulated by manually opening, throttling, or shutting globe valves.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

XI. AUXILIARY SYSTEMS AND MACHINERY

XI-1 Reactor Auxiliary Systems

XI-2 Facility Auxiliary Systems

XI-3 Waste Disposal

XI-4 Fire Protection System

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

XI-I. REACTOR AUXILIARY SYSTEMS

1. FUEL ELEMENT HANDLING

Fuel elements are grappled and locked to the fuel handling tool, raised out of the core, and transferred to the drop tube as shown in Figure XI-1-1. The fuel handling tool is a long handled hook tool that is operated manually through the reactor head elliptical ports. The fuel element is covered with water during all phases of transfer to provide adequate cooling and shielding.

After the element is placed in the canal-transfer-device insert, the transfer device is actuated and the element removed with a hook tool. The canal-transfer-device insert is designed to limit the number of fuel elements being transferred and to position the element in the transfer device at an elevation where it can be reached readily for attachment and release. Subsequent fuel handling operations require placing of fuel elements in storage racks for cooling, placing elements on the underwater saw table for removal of end frames, and placing of the cropped elements in another storage rack or directly into the fuel element transfer cask. The spent fuel element transfer cask is moved with a straddle carrier similar to the one used for handling ETR spent fuel element casks.

A pole-hoist tool is provided to aid in the removal of stuck fuel elements from the reactor core. The lower end of the pole-hoist tool may be positioned directly over a stuck component and a straight vertical force applied to dislodge a component.

2. CONTROL COMPONENT HANDLING

2.1 Safety Rod Assembly

Handling operations required for the safety rod housing are as follows:

- (1) Flux wire handling
- (2) Safety rod latch handling
- (3) Absorber plate handling
- (4) Flux trap filler handling
- (5) Safety rod and rack tube handling.

These handling operations are accomplished through the top-head elliptical ports with manually operated long handled tools. These tools are connected and locked to the component being handled; the component is then raised or turned to remove or unlatch the item. Items 1, 3, and 4 are raised out of the safety rod housing and transferred to the canal via the drop tube and canal transfer device. A transfer station, located adjacent to the fuel storage rack in the reactor vessel, is used when handling these items from some core positions. The component being

handled is placed in the transfer station when changing grappling tools. The component being transferred is placed in a canal-transfer-device insert that is sized to place the upper end of the item at the correct elevation for grappling or releasing in the reactor vessel and when transferred to the canal.

The safety rod latch tool is used to lock or unlock the absorber plate. No transfer operation is required with this tool.

The transfer of the safety rod and rack tube assembly (Item No. 5) is done with the safety rod and rack tube handling tool, the pole-hoist tool, and auxiliary hoisting equipment. Transfer is made in the vessel to the drop tube and canal transfer device. The assembly can also be transferred through the top closure plate, into the loop handling cask, and to the canal.

A safety rod and rack tube assembly is reassembled by visually aligning the rack with the drive pinion position and is lowered into place. Final rotational alignment is established during the positioning of the drive pinion to the rack.

2.2 Neck Shim and Regulating Rod Drive

Neck shim rods and regulating rods are removed with the aid of long handled tools used through the elliptical ports. The rod is grappled, raised, sheared, and passed through the drop tube into the canal transfer device. The remaining neck shim drive cable is pulled up through the top-head ports. New neck shim rod assemblies are installed through the top-head ports.

Connection to the drive and position adjustment is made in the neck shim rod drive room.

2.3 Outer Shim Control Handling

The gear box is transferred to the storage canal via the canal transfer device after removing the drive shaft, the coupling bolt, connecting spline shaft, and mounting bolts.

Outer shim drums are removed by first removing the gear box support beam with attached gear boxes. The drums are then lifted out of the beryllium reflector and lowered through the drop tube into the canal-transfer-device insert. Poison inserts are removed from the drums in the canal maintenance area with the use of special handling fixtures and tools. A minimum of seven feet of water, used as shielding, is required over the drum in the fixture. After replacement of poison inserts, the drum is transferred back through the canal transfer device to the reactor vessel. The drum is reinstalled and the beam reassembled. Correct position of the poison insert to the drive is established by turning the shim drive until the drum is stopped against either the full-in or the full-out stop on the beam. Fuel must be removed from the core in the area from which outer shim drums are to be removed. Fuel removal is necessary to permit raising of the flux trap baffle to allow for support beam removal.

3. INTERNALS HANDLING

3.1 Vibration Ties

The vibration tie housing is not removed or altered except for major reactor internals disassembly. Removal of the neck shim housing or the top-head closure plate requires removal of the vibration tie housing. Vibration tie support arms are swung against the reactor vessel wall to facilitate lateral transfer operations to the drop tube.

3.2 Neck Shim Housing Pressure Tube Locator Ties

The locator ties can be pivoted from the arm pivot bolt, or the supporting ring may be opened by releasing the tie bolt. It is necessary to swing the locator ties out of the way for some handling operations such as removal of safety rod absorber plate and flux trap filler handling. Actuation of the locator ties is accomplished with wrench and hook tools. All hardware on the locator ties is captured to prevent dropping in the reactor complex.

3.3 Beryllium Reflector

Beryllium reflector assemblies are grappled and raised through the top-head access ports or through the full opening top-head. Special shielding provisions are made using local shielding to protect the crane operator.

3.4 Aluminum Reflector

The aluminum reflector blocks are handled with a special purpose, long handled tool operated through the access ports. The blocks are transferred to the canal via the drop tube and the canal transfer device.

3.5 Neck Shim Rod Housing

Removal of the neck shim housing is a major disassembly task and requires prior removal of fuel, pressure tubes, pressure tube locator ties, pressure tube vibration tie assembly, flux trap baffle, neck shim, and regulating rods and sleeves. The neck shim rod housing is removed as a two piece assembly with the overhead crane operated through the access ports or with the top head and closure plate removed. Since the housing is withdrawn from the vessel, local shielding may be required along with personnel evacuation from the area. The housing will be placed in the storage canal for storage prior to final disposition or for disassembly and servicing.

3.6 Neck and Regulating Rod Sleeve

The sleeves can be removed from the neck shim rod housing with a special long handled tool after the pressure tube locator tie has been swung out of the way and the neck or regulating rod has been removed. The sleeve is raised out of the housing and transferred via the drop tube and canal transfer device to the canal.

3.7 The Flux Trap Baffle

The flux trap baffles require prior removal of pressure tube, flux wires, safety rod absorber plate, flux trap fillers, and safety rod guide and rack tube. The pressure tube vibration tie assembly must be raised and the pressure tube

locator ties swung out of the way. The flux trap baffle is transferred laterally to the drop tube and to the canal via the canal transfer device. A new flux trap baffle is transferred through the access ports and positioned in the core.

3.8 Drop Tube Cover

The drop tube cover is removed from the drop tube for transfer operations after the water level is lowered in the reactor vessel to the canal water level (elevation 100 feet). The cover latches are loosened and the cover is transferred to the cover storage rack with long handled tools through the access ports.

3.9 Reactor Internals Protective Covers

The protective covers are used to catch loose items in case they are dropped into the reactor vessel. The protective covers are of three types: one of solid plate, another with a series of 1/2-inch-diameter holes drilled through the cover plate, and the third with trapezoidal-shaped openings 9/16 x 1-11/16 inches. Covers may be loosened and removed through the drop tube with long handled tools operated manually through the reactor vessel access ports. New covers are lowered into the reactor vessel through the top-head access ports.

4. EXPERIMENT HANDLING

4.1 Loop Handling

Loop handling is divided into the following categories:

- (1) Test assembly handling
- (2) Pressure tube handling
- (3) Flow tube handling
- (4) Test assembly stuck in a flow tube.

New loop components are handled with the two-ton overhead crane. Irradiated loop components are handled with the Loop Handling Cask and an assortment of lifting tools and adapters.

4.11 Test Assembly Handling. An irradiated Loop Test Assembly is transferred wet, with the aid of the Loop Handling Cask, to a test-assembly can located in the cask transfer station for canal storage as shown in Figure XI-1-2. The cask is a shielded container with primary and secondary cooling systems capable of containing an irradiated test assembly for an extended period of time. The test assembly may be transferred with the cask via the cask transfer truck to other NRTS facilities.

4.12 Pressure Tube Handling. Irradiated Loop pressure tubes are transferred to the storage canal with Loop cask via the cask transfer station. The tube is cut in the subpile room and the tube raised until a seal plug is inserted in the reactor vessel bottom-head. The lower end of the pressure tube protrudes from the cask when positioned in the cask. When the cask is raised

FIG. XI-1-2 LOOP TEST ASSEMBLY HANDLING ARRANGEMENT.

and moved laterally to the storage canal, the reactor room area must be evacuated of personnel as dictated by health physics monitoring. New pressure tubes are lowered into the reactor vessel with the two-ton overhead crane.

The loop-puller assembly is used to provide a controlled push or pull to the pressure tube when installing or removing the tube.

4.13 Flow Tube Handling. An irradiated flow tube may be removed from a pressure tube with the Loop Handling Cask after removal of the Loop Test Assembly. The flow tube protrudes from the cask when in position. The reactor room area must be evacuated of personnel during raising and lateral transfer operations of the cask loaded with the flow tube. The cask is positioned on the cask transfer station, and the irradiated flow tube is lowered into the canal. It is not necessary to perform any cutting operations on the flow tube prior to removal from the in-pile tube.

4.14 Test Assembly Stuck in a Flow Tube. A flow tube with a test assembly stuck in it may be removed from the in-pile tube with the Loop Handling Cask. Sufficient time will be allowed after shutdown to permit test specimen residual heat generation to decay sufficiently for the test assembly to be transferred without endangering personnel or equipment. The transfer of the flow tube with a stuck test assembly is a dry transfer since the lower end of the flow tube protrudes from the cask. The transfer is similar to flow tube or pressure tube transfers in that the area must be evacuated during cask raising, transfer, and lowering operations. The assembly is the longest to be handled with the cask. If the test assembly becomes lodged in the flow tube after the test assembly has been partially removed from the flow tube, it may be necessary to cut off a section of the flow tube when the cask is raised to the full-up position. The cask is positioned on the cask transfer station, and the flow tube with stuck test assembly is lowered into the canal.

Gas loop handling will depend on final design considerations. However, methods and equipment for handling operations are expected to be similar to those outlined above.

4.2 Capsules

Capsule-type experiments are handled in the same manner as those performed in the MTR and ETR. Longhandled grappling tools, manually operated through access ports, are attached to capsule-containing baskets. The baskets are transferred to the canal via the drop tube.

Canal operations include viewing, disassembly of components, removal of flux wires, and other work. All experimental operations performed in the canal are under the direct supervision of the operating contractor. Procedures conform to standard practices developed over many years of test reactor operational experience in the MTR and ETR.

XI-2. FACILITY AUXILIARY SYSTEMS

1. PLANT AND INSTRUMENT AIR SYSTEMS

The existing MTR-ETR compressed air plant was expanded to supply 1760 cfm of compressed air at 165 psig to all three facilities for both plant and instrument air service by the installation of a 660 cfm compressor. All compressed air is dried to a dew point of minus 30°F. In the event of low instrument air pressure, all stored air is reserved for instrument air use by means of an automatic plant air shutoff valve. Plant air is delivered at 125 psig, and instrument air at 115 psig. Sufficient dried air is stored to maintain ten minutes of instrument operation in order to safely shut down all three reactors in the event of loss of all compressors. Manual shutoff valves are provided at all branch root connections from the air mains to isolate a leaking or ruptured line. Low instrument air pressure sounds alarms and signals ATR power reduction.

2. HIGH PRESSURE AIR SYSTEM

Two 250-psig compressors, one of which is a spare, provide 250 psig compressed air for starting the diesel engines, pressurization of the reactor primary coolant system surge tank, and HDW expansion tank. Sufficient 250-psig air is stored to provide for eight cold starts of a diesel engine.

3. DIESEL GENERATOR AUXILIARY SYSTEMS

3.1 Diesel Oil Storage

The existing diesel oil storage system was expanded to a total capacity of 135,000 gallons, adequate for 30 days operation for the entire facility, by the installation of a new 35,000-gallon tank. A 1000-gallon buried tank adjacent to the ATR building diesel generator area provides more than 12 hours of diesel operation in the event of supply line failure. In addition, a 10-gallon overhead tank in combination with a hand-driven priming pump is provided to assure an uninterrupted supply for a rapid diesel start. Diesel oil is filtered prior to storage in the 1000-gallon tank, and filtered again prior to injection into the engine.

3.2 Cooling System

The diesel-engine jacket-water system is provided with an expansion tank, an engine-driven jacket-water pump, and a motor-driven standby jacket-water pump. Potable water is used to fill the jacket-water system. Jacket-water heat is absorbed by the utility cooling-water (UCW) system.

3.3 Lube Oil System

The diesel engine lube oil system is provided with an engine-driven lube oil pump and a motor-driven, standby lube oil pump. Lube oil heat is absorbed by

the UCW system. The lube oil is filtered upstream of the engine. In addition, a continuous bypass filtering system is provided for the lube-oil sump tank.

3.4 Emergency Starting

Electric power is not required to start a diesel engine. The hand-priming pump supplies diesel oil; fire water is automatically supplied for jacket cooling; a large reserve of compressed starting air is maintained; and the engines are designed to start with residual lube oil retained on contact surfaces.

3.5 System Instrumentation

The diesel engines were furnished fully instrumented with alarms, interlocks, and safety devices to protect against misoperation or malfunction. The reactor reduction signal for loss of diesel power is actuated by undervoltage relays on the diesel bus.

3.51 Lubricating Oil System. A pressure switch starts the standby lubricating oil pump on less than normal lubricating oil pressure. On still lower pressure, a switch shuts down the diesel engine. An alarm rings on both the diesel and process control panels on low oil pressure.

On high oil temperature, an alarm sounds on both the diesel and process control panels.

3.52 Jacket Water System. High jacket-water temperature sounds an alarm on the diesel and process control panels.

3.53 Diesel Fuel Oil System. The diesel fuel oil day tank is furnished with a high- and low-level control switch and a high- and low-level alarm switch.

The level control switch actuates a solenoid valve controlling an on-off pneumatic operated valve on diesel fuel transfer to the day tank, and also starts and stops the transfer pump. On low level, the tank is refilled; on high level, the control valve closes and the transfer pump may be stopped. The transfer pump is also used to transfer oil to MTR and ETR and may be controlled from their systems.

The alarm switch actuates a high- and a low-level alarm on the process control panel.

Low pressure in the diesel oil fuel lines to the engines is sensed by a local pressure switch which actuates annunciator alarm on the diesel panel and process control panel.

3.54 Engine Speed. A tachometer indicator, with high alarm, is installed to annunciate overspeed on the diesel and process control panels.

3.55 Starting Air. Suitable controls and instrumentation are provided on the starting air storage tank to assure that sufficient air is always available for eight diesel engine starts.

4. PLANT HEATING AND PROCESS STEAM SYSTEM

The existing three-boiler steam plant has adequate capacity for the entire three-reactor facility including the ATR. Saturated steam at 125 psig is provided for building-space heating and miscellaneous process steam requirements. Existing fuel oil storage capacity is adequate for more than 30 days of operation.

5. POTABLE WATER SYSTEM

Well water is softened, blended with raw water, and chlorinated for personnel use. The existing potable water system has adequate capacity for the entire facility including the ATR.

XI-3. WASTE DISPOSAL

1. HOT WASTE

1.1 General Description

The flow diagram for the hot experimental waste system is shown in Figure XI-3-1. High-activity wastes from the experimental cubicles, the cubicle sample areas, and the reactor area flow by gravity through an internal tank strainer to the 1000-gallon hot experimental waste storage tank which is located in a shielded vault below the second basement floor.

The hot experimental waste tank is drained by two vertical, tank-mounted, 25-gpm pumps, one of which is a spare. One pump is supplied from commercial power and the other from diesel power. The pump motors are accessible at all times for maintenance and are located on the motor floor which is above and shielded from the hot waste tank vault.

1.2 Instrumentation and Sampling

The tank level is indicated and a high level actuates an alarm on the process control panel. The pumps are started and stopped by manual control from the process control panel.

The hot experimental waste pumps are equipped with a recirculation line which mixes the tank contents so that representative samples can be taken to determine the activity. The hot experimental waste is normally pumped to the existing underground MTR hot waste storage tanks. If the fluid has an activity level below that permitted in the warm waste system, the hot experimental waste can be pumped to the larger warm waste tank. This prevents overloading the hot waste disposal system with low-level wastes.

For pumping out the hot experimental waste tank residue, a portable pump is inserted through the connection normally occupied by the strainer.

The hot experimental waste tank pump discharge line can be drained back into the hot experimental waste tank to clear the discharge lines prior to removing the hot experimental waste tank pumps for maintenance.

The hot experimental waste tank vault has a floor sump with a level float switch which enunciates an alarm on high-water level.

1.3 Materials of Construction

The tank, pumps, piping, valves, and fittings in this system are austenitic stainless steel.

FIG. XI-3-1 WASTE HANDLING AND DISPOSAL SYSTEM.

2. WARM WASTE

2.1 General Description

The flow diagram for the warm waste system is shown in Figure XI-3-1. Low-activity waste from the rod access area sump, canal cleaning system, canal drains, bypass demineralizer drains, building drains, reactor top-head drain, degassing tank overflow drain, fission break monitor sample drain, and N-16 monitoring drains flow by gravity through an internal tank strainer into the 5000-gallon warm waste tank. The tank is located in a shielded vault below the second basement floor.

2.2 Instrumentation and Sampling

The warm waste tank is drained by two 200-gpm, vertical, tank-mounted pumps, one of which is a spare. A level switch automatically starts and stops both pumps. The pumps can also be started manually from the process control room where a high tank level is annunciated. The pump motors are accessible at all times and are located on the motor floor which is above and shielded from the warm waste tank. One pump is supplied from the commercial power system and the other from the diesel power system.

The warm waste tank pumps are equipped with a recirculation line which is used to mix the tank contents in order to assure representative samples for activity level determinations. MTR operators will administratively control discharge of ATR warm waste to limit activity in the leaching pond.

The warm waste is pumped to the 30-inch line from the MTR Process Water Building to the retention basin. The liquid is ultimately discharged to the leaching pond.

The warm waste tank strainer can be removed for cleaning or for pumping out the warm waste tank by the use of a portable pump.

The warm waste tank pump discharge line can be drained back into the warm waste tank to clear the discharge line if the warm waste sump tank pumps are to be removed for maintenance.

The warm experimental waste tank area has a floor sump, with a level float switch which annunciates an alarm on high water level.

2.3 Materials of Construction

The tank pumps are cast iron construction with steel column pipe. The piping and valves in the reactor building are stainless steel. Inside and outside buried lines which lead to the MTR-ETR area are ductile cast iron.

The primary coolant system flush water drain joins the warm waste sump tank drain for disposal to the MTR facilities. The primary coolant flush drain piping is austenitic stainless steel inside the reactor building and is ductile cast iron outside the building.

3. COLD WASTE

Cold wastes from the cooling tower blowdown and basin drain, as well as from the cubicle air-handling equipment, flow by gravity to manhole 22C and then to the disposal well. Cooling tower overflow flows by gravity to the ATR area discharge ditches.

4. SANITARY WASTE

Discharges from the ATR sanitary facilities flow by gravity to the existing MTR sewage treatment plant.

5. SOLID WASTE

Solid radioactive wastes are disposed of in the existing NRTS burial grounds using the same type of facility hauling and burial equipment as used for the MTR-ETR.

6. GASEOUS WASTE

The hot experimental waste tank, the warm waste tank, all drain lines emptying to these tanks, and the two waste tank vaults are maintained under a negative pressure by means of a supplementary exhaust fan which discharges to the main exhaust system. All drain lines are orificed to assure a negative pressure in the waste drain systems. This provides positive removal of radioactive gases present in the liquid wastes. These drain and vent lines have no water seals or other restrictions which could act as a collection point for radioactive crud.

XI-4. FIRE PROTECTION SYSTEM

The MTR-ETR-ATR facility's fire protection system is served by a common water supply and storage system which is fed from deepwells. The NRTS area has a natural underground reservoir of water having an estimated lateral flow of 300,000,000 gallons per day.

The water supplied by the two existing deepwells was supplemented by Number 4 deepwell pump of 2000 gpm capacity. This addition makes the total capacity 9000 gpm.

The deepwell pumps discharge into three existing ground level, water storage tanks having a combined capacity of 1,500,000 gallons. In addition, an existing 150,000 gallon overhead water storage tank pressurizes the fire-water system to a minimum of 78 psig. Two existing 5100 gpm fire-water pumps, one motor driven and one gasoline engine driven, supply and pressurize fire system whenever the fire-water header pressure falls below 55 psig.

The ATR area is encircled by a buried fire-water loop of 12-inch ductile iron pipe, with hydrants spaced not more than 300 feet apart. The ATR cooling tower is protected by an automatic deluge fire system, two hose stations, one at each end of the cooling tower, and two 600 gpm hydrants. A total fire-water capacity of 4500 gpm is required which is available with one fire pump in operation. Any two of the three deepwells and either one of the two fire pumps have a capability in excess of the fire fighting requirements of the ATR cooling tower which is the largest of the facility's cooling towers. Thus, a continuous supply of fire water is available, permitting the storage tanks to be kept in reserve for other emergencies. During a protracted fire, the cooling tower makeup flows are cut off to conserve water.

The fire protection system, both in the yard area and the reactor building, is designed in conformity with IDO Engineering Standards (Section I-37), the applicable national and local codes, and the standards listed in IDO Engineering Standards, which include IDO-12008.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

XII. COUPLING BETWEEN REACTORS IN THE MTR-ETR-ATR AREA

- XII-1 General Considerations
- XII-2 Coupling by Direct Radiation During Normal Operation
- XII-3 Meteorological Coupling During Normal Operation
- XII-4 Coupling Through Common Facilities
- XII-5 Summary
- XII-6 Reference

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

XII-1. GENERAL CONSIDERATIONS

The possible interactions between the ATR and other facilities in the Test Reactor Area (TRA) which includes the MTR-ETR-ATR complex have been investigated in the following categories:

Coupling by direct radiation during normal operation

Meteorological coupling during normal operation

Coupling through common facilities.

(Coupling following an accident is analyzed in Section XVII).

XII-2. COUPLING BY DIRECT RADIATION DURING NORMAL OPERATION

The direct radiation from one reactor facility to the other at the MTR-ETR site has resulted in shutdown when the area radiation monitors of one plant responded to sources in the other. This has been an operational inconvenience. The relatively greater separation of the ATR from the other TRA facilities will reduce possible coupling by direct radiation.

XII-3. METEOROLOGICAL COUPLING DURING NORMAL OPERATION

Meteorological coupling during normal operation can be caused by stack effluents being carried back to ground level and into the reactor buildings. This can produce higher than normal radiation levels which may require the evacuation of the affected buildings. Some difficulty of this sort has been experienced in the operation of MTR and ETR partly because of the location of the ETR building in relation to the MTR and ETR stacks and partly due to the relatively high stack source strengths from the MTR and ETR as a consequence of their experimental requirements. Details of MTR-ETR meteorological coupling experience have been reported [1].

The most frequent winds associated with fumigation and other unfavorable meteorological conditions are about equally divided between northeast and southwest. Therefore, the ATR building and its stack have been located to the northwest of MTR-ETR. Winds in a direction to carry effluent from the MTR and ETR stacks to the ATR building and the reverse occur less than five percent of the time. In addition, the ATR stack is located such that areas in the most frequent wind direction (NE and SW) are unoccupied.

The principal sources of MTR stack gas activity during normal operation have been Argon-41 from the air that cools the reactor graphite, the off-gas from the process water, and the off-gas from experiment primary coolant and vents, including an open cycle air cooled experiment. At ETR, the same sources exist except for graphite cooling air. Off-gas activity from the ATR is expected to be substantially the same as at ETR except that open cycle air cooled experiments are not planned for the ATR. The ATR has no air cooled reflector, and the Argon-41 resulting from its primary shield cooling arrangement is small, compared to off-gas from the primary coolant. In view of these factors, the ATR stack effluent activity is expected to be lower than that experienced at ETR.

The relatively low ATR effluent activity combined with the favorable ATR stack location makes the meteorological coupling between ATR and the existing reactors less than the coupling between the MTR and ETR.

XII-4. COUPLING THROUGH COMMON FACILITIES

The MTR, ETR, and ATR share a number of common auxiliary systems and facilities. Since the initial installation of the MTR, these systems have been augmented with parallel equipment and transfer facilities to satisfy the requirements and to ensure dependable service to the MTR, ETR, and ATR. The ATR is designed so that a failure of an interconnected system from causes associated with the ATR will result in little or no disturbance to the other reactors. However, a gross failure of any system if not corrected within a reasonable time will cause all reactors to shutdown. All reactors are designed to be shutdown safely if the failure of any system is not corrected in time. These failures would not depend on the proximity or coupling of the ATR.

Systems used by the ATR that are common to all reactors in the complex are divided into two parts: those indirectly affecting reactor operations and those directly affecting reactor operations.

1. COMMON SYSTEMS INDIRECTLY AFFECTING REACTOR OPERATIONS

The three reactors are served by the following common systems whose failure at the ATR would indirectly affect reactor operations.

Chemical systems

Diesel oil storage

Potable water system

Heating steam and condensate

Liquid waste disposal

Communications

The failure at the ATR of one of the above systems would not immediately effect any reactor operations, but might over a period of time, if uncorrected, force the operators to shut down the ATR and in some cases the other reactors. In the case of all pipe ruptures, adequate valving is available to isolate the severed pipe.

1.1 Chemical Storage

All chemicals handled at the ATR are from individually sealed containers or tanks with the exception of the sulfuric acid which is pumped from one of two MTR acid storage tanks to the Cooling Tower Pump House. An accident involving the ATR chemical systems would provide only a local problem for the ATR operators, but it would not necessarily affect any reactor operation.

The effect of chlorine gas escaping from one of the two storage bottles in the ATR Cooling Tower Pump House, which is located north of the ATR Reactor Building and northwest of the MTR-ETR area, has been analyzed, and it was concluded that this accident will not endanger any personnel in the site. Meteorological coupling data, the dilution effects of the atmosphere, and the location of the bottles in a separate enclosed room with only one door facing west was considered in reaching the above conclusion.

1.2 Diesel Oil Storage

A rupture of the diesel oil supply line to the day tank, ATR-775, would not effect operation at the MTR or ETR since each facility has a sufficient supply of diesel oil in the day tanks (minimum tank capacity of eight hours supply) to ensure continued operation. The transfer pumps located in MTR-627 building would still be capable of delivering diesel oil to the other day tanks after the ruptured line is valved off by the operators. It is reasonable to assume, considering the time available from the diesel oil storage in day tanks, that the operators can valve off any line break in a few hours. The other plants would not be affected by a line break at the ATR. A rupture of the diesel fuel oil line to the ATR diesel engines would force the shutdown of the ATR but would not affect the other reactors.

1.3 Potable Water System

The loss of potable water to the ATR will not affect the operations of the MTR and the ETR reactors but will inconvenience the operators. However, since potable water is used for the ATR diesel-engine jacket-water makeup, the loss of this service over a period of time will cause the shutdown of the ATR diesel engine generators and, thus, the ATR reactor. Similarly, if potable water is used for makeup service in the MTR and ETR auxiliaries, then these plants may also be similarly affected if potable water is lost site wide.

1.4 Heating Steam and Condensate

The steam plant is capable of supplying the requirements of all three reactor plants, and the loss of any supply or return line to or from the ATR will not effect the MTR or ETR plants. The loss of steam supply or condensate return lines to the ATR will inconvenience the operators due to the loss of the heating system, but will not affect the operation of the ATR reactor.

1.5 Liquid Waste Disposal

Interruption of Liquid Waste Disposal system in the ATR will not affect the operation of the MTR and ETR in any way since the waste storage and disposal systems are designed to handle all three plants. On loss of the warm waste system, the ATR operators can also take action to restrict any noncritical warm drains including floor drains which are fitted with covers. However, the prolonged loss of the use of the ATR warm waste system will affect the operation of the ATR reactor since the gland seals and N-16 monitor water, which are necessary for continued operation of the ATR, drain to the warm waste tank.

1.6 Communications

The ATR communication facilities consist of three (3) separate systems. Of these, only the telephone and code call systems are interconnected with the MTR and ETR systems.

The telephone system is an extension of the Intermountain Telephone facility which serves the NRTS. This facility includes automatic switching equipment which serves all of the ATR desk and wall phones. All telephone and code call systems are connected to a common terminal rack in the ATR area. Any malfunction of this ATR installation will not affect service in the MTR and ETR areas.

The paging and intercommunication system consisting of speakers, handsets, amplifiers, and an intercommunication switching panel on the reactor console is not interconnected with the MTR or ETR. With this system it is possible for any paging station to initiate an ATR plant-wide page and then carry on a public or private (party line) conversation with the person paged. In addition, the ATR reactor operator at the console has intercommunication connection with a limited number of locations, such as experiment consoles, so that he can issue instructions, check instrument calibrations, etc, without the use of plantwide paging system.

The ATR dc telephone system operating from rectified 120 volt, 60 cycle ac with a central patchboard but without signaling capability is not interconnected with the MTR or ETR. With this system it is possible to provide prearranged service between any two (2) stations. This type of communication is used primarily for prolonged communications such as trouble shooting or instrument calibration.

2. SYSTEMS DIRECTLY AFFECTING REACTOR OPERATION

The three reactors are served by the following common systems whose failure at the ATR could directly affect reactor operations.

Commercial electric power systems

Instrument and plant air systems

Raw-water system

Fire-water system

Demineralized water system

Evacuation and fire alarm

The failure at the ATR of one of the above systems will cause the shutdown of the ATR either by a scram or an immediate action by the reactor operators. Failure at the ATR will not affect MTR-ETR operation if timely corrective action is taken by the operators. In the case of all pipe ruptures, adequate valving is available to isolate the severed pipe.

2.1 Commercial Electric Power System

All three reactors share a common source of commercial power which is used for driving the primary and secondary pumps. Loss of commercial power to the site will shut down the three plants simultaneously. The proximity of the

reactors does not affect the total number of such failures. Each reactor plant is provided with emergency power from individual diesel generator systems which are not interconnected with each other or normally interconnected with the commercial systems.

The 13.8 kV commercial power feeders for the ATR and ETR are supplied from common transformers. Each feeder to the ATR and ETR has its own circuit breakers which isolate branch circuit faults. In addition, the two ATR feeders are connected through a tie breaker. The two ETR feeders are similarly connected. In the event of a fault in any feeder, the appropriate circuit breaker will open. The single feeder remaining for this system is still operative and, depending on the location of the fault, may feed part of all of its plant load. In the event that the circuit breaker does not open on the faulted system, the 132 kV supply to the affected transformer will be interrupted. Full load can still be furnished to the system through the tie breakers from the remaining 132 kV supply feeder. Thus, a fault on the ATR commercial power supply system can be isolated so that operations of the other reactors are essentially unaffected. Similarly a fault on the ETR commercial supply system can be isolated so that operation of the ATR or MTR is unaffected.

2.2 Instrument and Plant Air System

The MTR, ETR, and ATR have a common air supply system from common air receivers and dry-type air compressors, some of which are powered from the diesel system and the rest from the commercial system. This arrangement ensures dependable and adequate air supply in the event either power system fails. The air system consisting of both 125 psig plant air and 115 psig instrument air is supplied through pressure reducing valves. A rupture of the plant air line to the ATR or any other plant cannot affect any reactor since the plant air system is equipped with an air shutoff valve which shuts off all plant air when the system pressure from the receivers falls to 100 psig. Thus, the remaining air is conserved for instrument use.

The severing of 1-1/2 inch or larger instrument air pipe will cause the instrument air system to depressurize over a period of time. At 100 psig the plant air will be shut off as above and alarms sounded. In the event that the leak cannot be isolated by the operators in time, the ATR reactor will scram when the instrument air pressure falls to 60 psig.

Similarly, the ETR and MTR will also be shut down on low instrument air pressure.

2.3 Raw Water System

Raw water is supplied to each of the reactor plants from the 150,000 gallon overhead storage tank through separate lines. Although it is very unlikely, complete rupture of the six-inch main supply line to the ATR would release approximately 1500 gpm from the raw water system. Such a rupture would interrupt the flow of cooling water to the ATR primary coolant and pressurizing pumps forcing the shutdown of the ATR reactor.

On low overhead tank level, MTR cooling tower makeup would be automatically restricted and all raw water pumps discharge flow routed to the overhead tank. Any two out of four raw water pumps can provide a minimum of 1700 gpm to the overhead tank. Therefore, the loss of raw water at the ATR will not affect operations at the MTR or ETR.

2.4 Fire-Water System

The MTR, ETR, and ATR share a common fire-water system arranged in loops. The only major fire hazards in the area are the cooling towers, which are not likely to catch fire when wet and in use. At present, it is intended to have only one reactor and thus one cooling tower shut down at a time. The largest cooling tower (ATR) is protected by a deluge system and hydrants, with a combined fire fighting requirement of 4500 gpm which can be supplied from any one of the two 5100 gpm fire pumps.

The largest fire-water supply pipe to the ATR building is six inches, which is also the size of the supply line to each hydrant. The rest of the system is buried. On rupturing, this six-inch pipe will discharge approximately 1500 gpm which can be resupplied by any one of the two 5100 gpm fire pumps; 2000 gpm of fire water is also required by the reactor for emergency cooling.

It is considered incredible that a cooling tower would catch fire, a six-inch pipe or hydrant would break or rupture completely, and a reactor require emergency cooling water, all at the same time. However, one of the centrifugal-type fire pumps or any two of three deepwell (deepwell pump Number 2 excepted) pumps, one of which was installed with the ATR, can adequately supply fire water for any combination of two of the three above requirements. However, both fire pumps or all three deepwell pumps could supply enough water for the three above requirements. There is also a reserve supply of water that can be used for fire fighting purposes in the three 500,000-gallon raw water storage tanks, and the 150,000 gallon overhead storage tank. It is also considered incredible that two reactors, two cooling towers or two pipe breaks or any such combination in the area would require fire water at the same time. In addition, there are shutoff valves available to isolate any pipe or hydrant break and reduce nonessential raw water flows, such as cooling tower makeup, at the operators discretion. Thus, a fire at the ATR will not effect reactor operations at the other plants nor will a fire at other plants affect reactor operations at the ATR.

2.5 Demineralized Water System

The demineralized water supply to the ATR comes from one of two common 500 gpm transfer pumps, the other being standby, taking suction from the two 100,000 gallon storage tanks. The demineralized water system was supplemented during the construction of the ATR to provide an adequate and reliable supply of demineralized water for all three plants. It is considered incredible that the complete rupture of a 10-inch stainless steel pipe which has no portion exposed, except overhead in buildings, could occur. More probable is the severing of a smaller pipe in an accessible location such as a 1-1/2-inch canal purge feeder at the canal parapet wall. Such a break would not deprive the ATR nor the other plants of demineralized water for essential services; shutoff valves are available to allow operators to quickly isolate a ruptured pipe. In the event a valve is closed through misoperation so that the supply of demineralized water to the ATR is interrupted, thus shutting off gland seal water and primary coolant system makeup, the ATR reactor would have to be shut down by the operators; 2000 gpm of fire water is available to cool the ATR reactor core in the unlikely event that the interruption of all demineralized water occurs at the same time as a gross leak occurs in the primary coolant system.

2.6 Evacuation and Fire Alarms

The ATR Evacuation Alarm system is interconnected with and is a duplicate of the MTR and ETR systems. Each reactor operator can signal an evacuation or area alert either on a local or site basis. Malfunctions would affect reactor operations as required by the site safety regulations.

The ATR Fire Alarm system is interconnected with and is a duplicate (coded signal) of the systems used in the MTR and ETR. All alarms or supervisory signals in any area are sounded in all areas and at Central Facilities fire headquarters. Malfunctions would affect reactor operations as required by the site safety regulations.

XII-5. SUMMARY

The ATR building and stack have been located to use prevailing wind patterns to minimize releasing or receiving radioactive airborne material to and from the MTR-ETR area. The ATR distance from the MTR-ETR area makes direct radiation hazards between areas a remote possibility. Supplementary equipment installed in auxiliary systems common to the three reactors has increased system reliability. New and existing piping systems have been designed with shutoff protection in the event of localized failure. Electrical systems are designed to isolate faulted areas.

Based on the above factors, it is concluded that the addition of the ATR does not represent a significant increase in hazards to the TRA area during normal operation. Failures in systems common to all reactors do not represent a safety problem to any of the reactors, but may require one or more reactor shutdowns.

XII-6. REFERENCE

1. D. R. deBoisblanc et al, Hazards Evaluation of Proposed Site for ETR II, Phillips Petroleum Company, IDO-16623 (August 1960).

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

XIII. EMERGENCY EVACUATION PROCEDURES

XIII-1 Evacuation

XIII-2 ATR Evacuation Plan

XIII-3 Reference

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

XIII-1. EVACUATION

1. GENERAL

A large release of radioactive gases from any of the facilities at NRTS could require the evacuation of one or more on-site areas and, depending on the magnitude, one or more off-site localities. The evacuation of areas other than the site of the release has not been necessary in the past. There have been releases at MTR and ETR which have required the evacuation of either or both MTR and ETR. A large release at ATR could require the evacuation of all three reactors. The NRTS has written procedures for carrying out these evacuations [1]. A line of command has been established to authorize and implement the procedures as required to fit the actual circumstances. Alternate supervision has also been designated in case the primary leader is disabled or absent.

2. EVACUATION OF THE MTR-ETR-ATR SITE

The existing MTR-ETR site has a written shutdown procedure for the reactors and the experiments and a written evacuation procedure. The same type of procedures are being prepared for the ATR. Evacuation routes are established and the responsibilities of evacuation wardens are defined.

When the personnel reach the evacuation area they board buses for evacuation. During the night, Sunday, and holiday shifts, two evacuation buses are parked in the parking lot for emergency use. During the normal day, shift buses are sent from Central Facilities to remove the personnel. During a Chemical Processing Plant incident the time for complete daytime evacuation by bus was 30 minutes from the time the alarm was sounded. MTR-ETR evacuations were completed in less than 20 minutes.

An incident at ATR may require evacuation of MTR and ETR. However, such evacuation will not contribute to the overall hazard because (a) emergency procedures have been established for the safe shutdown of ETR and MTR prior to evacuation, (b) the design of these plants is such that each reactor can be scrammed immediately, (c) the auxiliary and experiment systems are automatically shut down, and (d) emergency coolant is automatically provided for extended periods without operators in attendance.

3. EVACUATION OF OTHER NRTS SITES

Each NRTS site has an emergency evacuation procedure. This procedure would be set into operation upon notification of the supervisor of the endangered areas by the supervisor of the area where the accident occurred. This notification could be by telephone since all areas are connected to the NRTS telephone system. There is also an NRTS radio communication system. Notification of hazardous events is processed through the Warning Communication Center (WCC) and secode radio net.

The NRTS transportation plan covers the dispatching of buses to the various areas for evacuation. The plan includes stationing buses at all times at the more distant areas such as TAN, Spert, ANL, and the GCRE, ML-1 areas. The areas closer to the Central Facilities Areas have standby buses during night, Sunday, and holiday shifts, but have buses sent to them from CFA to evacuate the large normal day shifts. Table XIII-1-1 is an excerpt from the IDO Emergency and Disaster Handbook [1] describing the disposition of buses. All buses are normally parked at CFA except where specifically noted.

TABLE XIII-1-1

TRANSPORTATION PLAN

AEC and their Operating Contractor employees will be furnished evacuation transportation by the Phillips Petroleum Company, operators of the AEC-IDO transportation system. Construction Contractors will normally depend on company vehicles, personal vehicles, and charter buses for evacuation.

TAN Area

A minimum of five buses is assigned to this area to transport day shift workers. They are to be operated by Phillips Petroleum Company employees. During all other hours, three Metro buses and available sedans in the area will be used to evacuate employees.

NRF Area

Twenty-three buses will be dispatched to this area to evacuate day shift personnel. Three buses are assigned permanently on standby during all hours.

TRA Area

Twelve buses will be dispatched to this area. These plus two standby buses will be used to evacuate personnel during the day shift. During all other hours, the two standby vehicles together with available sedans will be used to evacuate personnel.

ICPP Area

Four buses will be dispatched to this area to evacuate day shift employees. Plant assigned vehicles will be used to evacuate personnel during all other hours.

CFA Area

Eleven buses will be dispatched to evacuate day shift workers. During all other hours, available sedans will be used for evacuation.

Spert Area

One standby bus plus plant assigned vehicles will be used to evacuate personnel during all hours.

TABLE XIII-1-1 (Cont.)

OMRE Area

This plant has been deactivated and will not require vehicles except during special visits or during dismantling work.

ANL Areas

West Area -- Two buses are available at this area to evacuate personnel during the day shift. The full-time, standby bus will be used during all other hours.

East Area -- Four buses are available in this area to evacuate day shift personnel. During all other hours, plant assigned vehicles will be used.

GCRE, ML-1 Areas

One bus is permanently assigned to the GCRE area. It will be used to evacuate personnel at this area and the ML-1 area during all hours. If evacuation is south on Fillmore Boulevard, the SL-1, ARA Hot Cell and Support Facilities personnel may also use the bus although one bus is assigned to these areas.

Evacuation Routes

The map of the NRTS, Figure II-1-2, shows that in case of an accident at ATR, it is possible to evacuate the various NRTS sites along roads that lead away from ATR. This arrangement of roads permits evacuees to travel to areas of safety without having to pass through the high radiation level area which would surround ATR after a large release of fission products.

4. EVACUATION OF OFF-SITE AREAS

The distance between Idaho Falls and NRTS is so great that it would never require evacuation as a result of an accident.

There is no written plan for evacuating closer communities. However, if one were threatened by the consequences of an accident the occupants would be notified by telephone, radio broadcast, and direct contact. The people affected would be evacuated in their own cars. The problem of evacuation of the towns around NRTS is not an insurmountable one since these communities have low populations.

XIII-2. ATR EVACUATION PLAN

A comprehensive ATR Evacuation plan, similar to the one in practice at ETR, will be complete together with personnel training prior to ATR start-up. The ETR plan is shown below.

1. ETR EVACUATION PLAN

The purpose of this plan is to establish procedures for conducting an orderly evacuation from the ETR area should conditions require such action.

This plan considers specific procedures to be followed in the ETR area and is to be used in conjunction with Part I, Basic Plan, Phillips Company Emergency Action Plans for the NRTS.

It is important to remember that circumstances surrounding any evacuation may alter specific procedures to be followed. Modifications to these procedures may be made by the responsible supervisor as conditions require. If in doubt, evacuate to the main gate using normal evacuation routes.

I. EVACUATION ALARM SYSTEMS

Three separate alarm systems are available for initiating partial or area-wide evacuation of the ETR and MTR-ETR areas:

- A. ETR Building Paging System: For a condition requiring evacuation of only one part of the reactor building (reactor basement, console floor, etc) the building paging system together with warden vigilance will be used to control evacuation from the danger area.
- B. ETR Area Alarm System: For conditions requiring evacuation of the entire ETR area, 13 sirens installed throughout the ETR area will be sounded. These sirens are activated from a switch in the reactor control room (labeled ETR Area Evacuation) and will sound in the following locations:
 - Office Building, First Floor
 - Office Building, Basement (Amplifier Room and H and V Room)
 - Reactor Building, Main Floor
 - Reactor Building, Console Floors (North and South)
 - Reactor Building, Basement (East and West Ends)
 - Electrical Building, Main Floor (East and West Ends)
 - Electrical Building, Basement (Cable Vault)
 - HX-Compressor Building
 - Process Control Room
- C. MTR-ETR Area Alarm System: Should conditions originating at ETR require that the MTR as well as the ETR be evacuated, an area-wide siren alarm system can be activated from a switch in the reactor control room (labeled MTR-ETR Area Evacuation). This will also sound code 222 on the fire alarm system.

II. EVACUATION ALARM SIGNALS

- A. Signal for ALERT: An approximate three-minute continuous sounding of the sirens with nonvarying intensity. (Alarm switch handle in down position.) Stay inside.
- B. Signal for EVACUATION: A five-minute continuous sounding of sirens, 5 seconds ON, 5 seconds OFF, on an alternating intensity. (Alarm switch handle in up position.)

III. EVACUATION PROCEDURE

The nature and degree of the emergency will determine the extent of plant shutdown required and the speed with which the shutdown must be effected. The order to evacuate will normally be given by the duty Shift Supervisor or his delegated representative. The extent of evacuation and plant shutdown will be determined by the duty Shift Supervisor or higher authority. This will usually fall in one of three categories:

- A. Evacuation of nonessential personnel only.
- B. Evacuation of all personnel; planned plant shutdown.
- C. Evacuation of all personnel; crash plant shutdown.

In the absence of word to the contrary, all personnel should assume a crash plant shutdown.

Each of these three situations is discussed below with suggested plant shutdown measures. Changes may be justified as conditions warrant by direction of the supervisor in charge.

EVACUATION OF NONESSENTIAL PERSONNEL ONLY

This type of evacuation is in order when conditions are not sufficiently severe to require shutdown of the plant.

The plant continues to operate with essential personnel only, wearing prescribed protective equipment.

EVACUATION OF ALL PERSONNEL; PLANNED SHUTDOWN

Shutdown of the plant will proceed as follows:

1. Reactor: By direction of the supervisor in charge, the reactor will be manually scrammed. The control console will be locked and the key delivered to the Shift Supervisor.
2. Reactor Experiments: Following shutdown of the reactor, the major loops will be attended according to the general requirements below. If the plant is already in a shutdown state at the time of evacuation, no action may be necessary. Specific instructions, if required, will be posted at loop control centers.

Water Loops

- (a) Reduce temperature. Shut down all line heating devices (line heaters, Dowtherm heaters).
- (b) Reset loop pressure controllers to approximately 700 psig.
- (c) Leave a minimum of one loop pump operating.

Gas Loops

- (a) Reduce temperature by shutdown of all preheat or other heating devices.
- (b) Maintain normal conditions of pressure and flow unless otherwise directed by the duty supervisor.

3. Process Water Facilities: After the reactor is shut down, process water equipment will be attended as follows:

- (a) Primary Cooling Pumps: Normally, all primary operating pumps will be left running. In the event the pumps are shutdown for some reason, be sure emergency power is supplying normal shutdown flow (approximately 2000 gpm).
- (b) Pressurizing Pumps: The operating HDW and the reactor pressurizing pumps will be left running.
- (c) Secondary Cooling Pumps: Shut down all main pumps. Two UCW pumps will be left running.
- (d) Cooling Tower Fans: Shut down all fans. During cold weather when icing will occur, switch tower flow through the lower jets.
- (e) HDW Circulating Pump: The operating pump will be left running.
- (f) Chemical Systems: These units (for Dianodic, chlorine, etc) should be shut down.

4. Experimental Air Facilities: After the reactor is shut down, this equipment will be attended as follows:

- (a) Oil Fired Preheaters: This equipment must be shut down. Turn the heater "Start" toggle switches to the OFF position.
- (b) Clark Compressors: These units (unless essential to sample cooling, or to MTR) will be shut down and emergency flow (Sutorbilt) established if loop conditions require.

Shutdown of the compressors will be by direction of the Shift Supervisor or Foreman.

5. Reactor Service Facilities

- (a) Electrical: All operating MG sets and the operating diesel power unit will be left ON. However, the diesel unit requires periodic attention. If the evacuation is prolonged, temporary reentry should be made (if feasible) and the machine shut down when plant afterheat problems are no longer pertinent.
- (b) Heating and Ventilating: Except to the extent it is affected by operation of the control room "panic" buttons, this system will normally remain operating.
- (c) Waste Systems: The warm and hot waste tanks can be left in the same condition as at time of shutdown, providing that the warm tank is valved to the retention basin and in "Auto" control.

EVACUATION OF ALL PERSONNEL; CRASH PLANT SHUTDOWN

Shutdown of the plant will proceed as follows:

- 1. Reactor: By direction of the supervisor in charge, the reactor will be manually scrammed. The control console will be locked and, at the first opportunity, the key will be delivered to the Shift Supervisor. Both "panic" buttons will be depressed.
- 2. Reactor Experiments: In most cases, little or no attention will be given to operating loops, etc. Special instructions, if mandatory, will be conspicuously posted at individual experiments.
- 3. Process Water Facilities: In the case of immediate plant shutdown and evacuation, this equipment will normally be attended as follows:
 - (a) Primary Cooling Pumps: These pumps will remain operating.
 - (b) Pressurizing Pumps: Leave running.
 - (c) Secondary Cooling Pumps: Leave running.
 - (d) Cooling Tower Fans: Shut down fans on way out if this can be accomplished with no significant delay in evacuating.
 - (e) HDW Circulating Pumps: Leave running.
 - (f) Chemical Systems: Leave as is at time of evacuation, preferably shut down.

4. Experimental Air Facilities: After reactor shutdown, attend this equipment as follows:

- (a) Oil-Fired Preheaters: This equipment must be shut down.
- (b) Clark Compressors: Following reactor scram, shut down these units using emergency stop buttons in Process Control Room. (At such time that fueled, air cooled experiments are again programmed into the MTR or ETR, this instruction may need revision.)

5. Reactor Service Facilities

- (a) Electrical: No attention will normally be given this equipment. All operating MG sets and the operating diesel power unit will be left ON.
- (b) Heating and Ventilating: Except as affected by operation of the control room "panic" buttons, this system will remain operating and require no attention.
- (c) Waste Systems: No attention will be given these systems.

IV. EVACUATION ROUTES AND PLANT EXITS

Evacuation routes from all ETR building areas are posted in strategic locations for the purpose of indicating the most direct exit from the building. These routes should be followed by evacuating personnel except in such obvious instances wherein the route would lead toward (rather than from) the danger area. Names of the regular wardens for each particular area are posted on the evacuation route signs.

All evacuations should be to the main west gate, unless there are obvious reasons for using other exits. Several emergency exits, located in the perimeter fences of the MTR-ETR area, are available for special conditions. (Refer to Standard Practice No. 10.220, MTR-ETR Emergency Action Plan, for further details.)

In general, always avoid the immediate disaster area and move upwind from contaminated areas, if possible. Every effort will be made by responsible supervision to give special guidance, if any is required, while the evacuation is in progress.

Evacuation should proceed to the MTR-ETR parking lot. Supervisors should report any missing personnel to the supervisor in charge. Further instructions will be issued from this point.

V. RESPONSIBILITIES

A. Personnel Designated to Declare Emergencies: Authority to declare an emergency and operate alarms is delegated to the MTR-ETR Plant Manager, the ETR Superintendent of Operations, or Assistant Superintendent, the Operations Shift Supervisor on duty, the Operations Shift Foreman, and the responsible Health Physicist on duty (radiation emergency only).

B. Warden Responsibilities: Evacuation wardens will be designated for each of the following building areas, and the names of these wardens plainly posted in the area concerned:

Reactor Building Main Floor and Electrical Building
Reactor Building Console and Basement Floors
Heat Exchanger-Compressor Building and Cooling Tower Pump-house Office Building.

It shall be the responsibility of the wardens to know thoroughly all details of the evacuation plan and to supply such checking and direction as may be required to see that the evacuation in their particular area proceeds smoothly and in accordance with all provisions of the plan.

It will be the responsibility of Health and Safety to keep an active list of wardens for all areas, to see that their names are properly posted, and to review the evacuation procedure frequently enough with the wardens to ensure that they are thoroughly familiar with all the relevant details.

The foregoing warden procedure applies only to regular daytime working hours. During night shifts, weekends and holidays, it will be the responsibility of the Operations Shift Supervisor, with the assistance of the Shift Foreman and other personnel designated by the Supervisor, to evacuate the plant and provide for the protection of classified material and other equipment as required in the evacuation plan.

C. Health and Safety Emergency Equipment: The duty Health Physicist as well as Operations Branch personnel should take portable survey instruments, respiratory equipment, and first-aid kits with them when evacuating, if feasible.

VI. DISASTER TEAM

A disaster team, composed of MTR-ETR Fire Brigade personnel, has been trained to assist during emergencies. Their primary function is to act as a reentry team, as directed by responsible supervision, for recovery of injured personnel.

Members of this team receive training in first-aid and use of emergency equipment. Upon total evacuation the team will assemble just inside the plant gate, if possible, and await orders from supervision. In other instances, they can be assembled by calling 4226 or 4306 (Health and Safety offices).

2. TRA EMERGENCY ACTION PLAN

The MTR-ETR Emergency Action Plan was modified to include the ATR. The modified Test Reactor Area (TRA) plan follows:

I. PURPOSE

To supplement the AED Basic Plan in regard to emergencies arising within or affecting the TRA.

II. EVACUATION

A. Evacuation Signal: An oscillating siren at five seconds on and five seconds off.

B. Action to be Taken:

(1) Upon receipt of an evacuation signal, all personnel will leave the Area and wait for the buses in the designated staging areas of the main parking lot.

(a) Unless directed otherwise, all personnel at MTR and ETR will evacuate through the main gate or the truck gate located west of the Hot Cell Building.

(b) Unless directed otherwise, all personnel at ATR will evacuate through the truck gate located directly south of the ATR Reactor Building.

(c) In cases where the use of the gates designated in (a) and (b) will be unsafe, other escape routes will be indicated by Health Physics, or Operations personnel, or evacuation wardens.

Emergency exits from the area are designated as "truck gates" or "turnstile".

Truck gates are approximately 20 feet wide. The gates may be center opening or side opening and are secured by soft iron chains and a padlock. Exit may be effected by cutting the chains or the padlocks with bolt cutters, which will be found nearby in bright yellow boxes.

Truck gates are located:

West perimeter fence opposite Hot Cell (MTR-657).

West perimeter fence opposite Storage and Receiving Building (MTR-662).

South perimeter fence by the Storage and Receiving Building (MTR-662).

East perimeter fence by MTR Working Reservoir (MTR-706).

North perimeter fence by ATR Reactor Mock-up Building (MTR-673).

Perimeter fence directly south of the ATR Reactor Building.

Turnstiles pass only one person at a time and are security alarmed. Exit is effected by walking through the turnstile in proper direction.

Turnstiles are located:

West perimeter fence opposite Hot Cell (MTR-657).

South perimeter fence near southeast corner of area.

North perimeter fence near northeast corner of area.

- (2) DO NOT GO INTO OR THROUGH ANY REACTOR BUILDING OR THE ALPHA CHEMISTRY BUILDING OR THE HOT CELL BUILDING DURING AN EVACUATION.
- (3) If directed to evacuate through exits other than those designated in (1) a and (1) b, proceed as rapidly as possible to the TRA access road and thence to its junction with road to NRF. Persons using the north exits or the east truck gate will proceed south along the outside of the east perimeter fence before attempting to reach the ATR access road. Further instructions and transportation will be available at the junction.
- (4) Two emergency evacuation buses will be used for evacuation during the dark shifts, Saturdays, Sundays, and holidays. The buses will be operated by ETR and MTR Operations personnel and normally will load at the TRA Main Gate and proceed to the CFA bus depot.
 - (a) The respective Operations organizations are responsible for keeping a full complement of drivers and assuring that the drivers are trained.
 - (b) Notification of evacuation conditions to WCC is the responsibility of the supervisor in charge during the evacuation.
 - (c) Personnel accounting is the responsibility of each supervisor during the evacuation.

III. ALERT

A. Alert Signal: A continuous sounding of the sirens with nonvarying intensity.

- (1) Upon receipt of an alert, employees are to take cover in the basement of the nearest major reactor building. The senior Operations supervisor present will be in charge until relieved in accordance with the Basic Plan.

IV. ALL CLEAR

Voice notification or personal contact.

V. SIREN CONTROLS

Either the "Evacuation" or "Alert" signal may be initiated from any control point. Control Points are located as follows:

A. MTR Control Room:

Total Area Evacuation

Reactor and Services Building Evacuation

Reactor Wing Building Evacuation

B. ETR Control Room:

Total Area Evacuation

Reactor and Office Building Evacuation

C. ATR Control Room:

Total Area Evacuation

Reactor and Office Building Evacuation

D. TRA Main Gate:

Total Area Evacuation

VI. SECODE NET

Emergency messages, concerning national or site emergency conditions, received on the radio secode net shall be immediately relayed to the assigned supervisor on duty. This supervisor shall make decisions deemed necessary and/or take appropriate action as advised by the message. The secode system can also be used for communication between user and WCC.

VII. EMERGENCY REENTRY TEAMS

A. Shift fire brigades are designated as Emergency Reentry Teams.

B. Members of the emergency reentry teams will report to their immediate supervisor in the appropriate staging area. They will then assemble at the staging area for Emergency Reentry Teams just within the plant gate and await further instructions.

VIII. RESPONSIBILITY FOR VISITORS

A. Visitors are defined as those persons who are not normally assigned to the NRTS.

B. Responsibility for visitors is assigned to their contacts in the TRA. All other personnel are expected to know the basic signals and the action to be taken.

IX. EMERGENCY EQUIPMENT

See Health and Safety SP 10.21.4.

X. TRAINING OF EMPLOYEES

Supervision will instruct employees in the requirement of this Plan and the Basic Plan at approximately six month intervals.

XIII-3. REFERENCE

1. Idaho Operations Office Emergency and Disaster Handbook, U. S. Atomic Energy Commission, Idaho Operations Office, Idaho.

XIV. INITIAL TESTS AND OPERATING PROCEDURES

XIV-1 Preoperational Systems Tests

XIV-2 Neutron Tests

XIV-3 Operating Procedures

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

XIV-1. PREOPERATIONAL SYSTEMS TESTS

The ATR equipment and hardware specifications require vendor testing in many cases to ensure satisfactory performance. In addition to these vendor tests, the installed (at the ATR) equipment is subjected to Component Continuity (CC) tests by the construction contractor. These tests and the results are reviewed by the AEC, the design contractor, and the operating contractor. The completed systems, after turnover to the operating contractor, are subjected to operational testing and performance evaluation by the operating contractor. These Systems Operations (SO) tests and results are reviewed by the AEC and design contractor.

1. COMPONENT CONTINUITY TESTS

A list of the major component continuity tests is presented below.

- CC-1 Special doors
- CC-3 Electrically operated cranes
- CC-4 Chain hoists
- CC-7 Building leak test
- CC-10 High-pressure air system
- CC-11 Primary system cleaning, flushing, and hydrostatic testing
- CC-12 Horizontal pump mounting, leveling, and alignment
- CC-13 Vertical pump mounting, leveling, and alignment
- CC-14 Air compressor mounting, leveling, and alignment
- CC-16 Diesel generator and auxiliary equipment
- CC-17 Demineralizers
- CC-18 Cooling tower
- CC-19 Cleaning and testing of secondary coolant piping and equipment
- CC-20 Cleaning and testing of plant general service systems
- CC-21 Process instruments
- CC-23 Electrostatic shielding
- CC-25 Transformers
- CC-26 Motors

- CC-27 Switchgear, motor control centers, and control panels
- CC-28 Motor-generator sets
- CC-29 Batteries
- CC-33 Data processing equipment
- CC-34 Motor operated valves
- CC-38 Heating and ventilating systems
- CC-40 Commercial power system
- CC-41 Diesel power system
- CC-42 Telemetering and supervisory control
- CC-43 Cathodic protection system
- CC-44 Lighting
- CC-45 Communications and alarm systems
- CC-101 Electrical check-out of reactor control, safety, and nuclear instrument systems
- CC-102 Handling sling
- CC-106 Reactor instrumentation drives

2. SYSTEMS OPERATION TEST

A list of the systems operation tests for the ATR is presented below. These tests include check-out and performance of the entire system including instrumentation and control response.

- SO-1 Primary coolant system
- SO-2 Secondary coolant system
- SO-3 High-pressure demineralized water system
- SO-4 Low-pressure demineralized water system
- SO-5 Utility cooling-water system
- SO-6 Gland seal water system
- SO-7 Liquid waste handling system
- SO-8 High-pressure air

SO-9 Cooling tower

SO-10 Underwater saw

SO-11 Raw-water system

SO-13 Bypass demineralizer system

SO-14 Cooling tower chemical treatment system

SO-16 Pumps horizontal

SO-17 Pumps vertical

SO-18 Instrument air system

SO-19 Compressed air dryer

SO-20 Telemetry and supervisory control

SO-21 Cathodic protection system

SO-23 Equipment load tests

SO-26 Radiation monitoring system

SO-27 Process instrumentation

SO-101 Reactor control, safety, and nuclear instrumentation system

SO-102 Reactor internals hydraulic testing

SO-107 Safety rod drives

SO-108 Neck shim drives

SO-109 Outer shim drives

SO-110 Storage space systems

SO-111 Reactor handling tools

SO-112 Canal transfer devices

XIV-2. NEUTRON TESTS

The ATR will be subjected to an extensive period of testing covering pre-loading hydraulic tests, critical loading, low-power physics tests, and approach to power and full-power tests. A report covering each test in detail will be prepared prior to loading fuel in the ATR. A preliminary outline of these tests is shown below. Additions and deletions to these tests will be based on results of research and development programs now in progress and ATRC results.

1. PRELOADING PROCEDURES AND TESTS

It is assumed that all CC and SO tests have been completed.

(1) Hydraulic Program -- A series of full core tests using instrumented dummy fuel elements will be made to determine if any hydraulic instabilities or undesirable interactions occur between fuel elements.

(2) Instrumentation Systems Response Tests -- Reactor and process control instrumentation will be subjected to open loop, frequency response tests to provide data to initially adjust control modes and provide a base for measuring the transfer function when the reactor is operational.

(3) Fuel Inspection Procedures -- The ATR fuel elements will be inspected to ensure dimensional specifications are met and gamma scanned for fuel content. Each element for the first core will be tested in the hydraulic facility at conditions of temperature and flow that are more severe than expected during operation. The entire core will be evaluated in the ATRC relative to reactivity and power profile measurements.

(4) Control and Safety Rod Poison Determination -- The ATR control and safety rod poison sections will be neutron source checked for absorption properties and worths measured in the ATRC.

(5) Detector Arrangement -- The detector arrangement to be used in the ATR for initial loading will be checked in the ATRC. The arrangement will minimize shadowing effects and ensure adequate sensitivity of the detectors.

2. CRITICALITY TESTS AND FULL CORE LOADING FOR THE ATR

(1) Critical Loading Determination -- The ATR will be loaded in the same way as is done in the ATRC. After source insertion, the detectors are calibrated and auxiliary detectors installed as required. Fuel loading is performed in a stepwise manner previously determined to be safe in the ATRC. The total reactivity of each step is limited and careful counting checks are made during each element insertion. After achieving a critical loading, a preliminary safety rod calibration will be made to ensure adequate shutdown capability for the full core loading.

(2) Full Core Loading -- Following rod calibration, stepwise loading of the full core is continued as above.

(3) Preliminary Power Calibration -- A preliminary power calibration will be made following the full core loading by foil measurements. These measurements will permit detector calibration and provide a comparison with ATRC results.

(4) Estimation of Shutdown k -- A preliminary estimation of shutdown k will be made to obtain a base for detailed rod calibrations.

3. POST NEUTRON, LOW POWER PHYSICS TESTS FOR THE ATR

(1) Preliminary Rod Calibrations -- Control rod intercomparisons will provide an approximation of total k and also yield a rough statistical weight map of the core.

(2) Determination of Excess k -- The excess k of the reactor will be determined by the distributed poison method.

(3) Detailed Flux Measurements -- Balanced Core -- Detailed core flux mapping will be performed with a balanced lobe power arrangement to provide data for predicting core and experiment heat transfer and performance.

(4) Detailed Rod Calibrations -- The worth of individual and ganged rod configurations will be determined for the safety, outer cylinder, and neck rods including interaction effects. Methods will be based on multiplication method, period measurements, pulsed neutron measurements, rod drop procedures, and intercalibration.

(5) Void Coefficients -- The importance of void effects (both positive and negative) on reactivity in the ATR requires detailed void measurements to be made in the various core areas.

(6) Fuel and Poison Statistical Weights -- The determination of the relative spacial distribution of the response to a small amount of poison and of a small fuel sample will be made. Comparing the values of the measurements with the corresponding quantity calculated using the adjoint flux will aid in setting up the core loading computational procedures to be used in future operations.

(7) Detailed Flux Maps -- Unbalanced Core Configuration -- Detailed maps of thermal flux cadmium ratios and fast neutron flux will be prepared for the ATR with the lobes at maximum allowable unbalance (40-50-60-MW power split).

(8) ATR-ATRC Intercalibration Tests -- Detailed measurements of flux, power, and other parameters will be made in the ATR and ATRC to establish equivalency.

(9) Isothermal Temperature Coefficient -- The temperature coefficient provides the major inherent safety action for ATR excursions and its value is therefore of marked importance.

(10) Reactor Safety Stability Studies -- Measurements of β/ℓ , interlobe kinetic behavior, and transfer function will be made to check calculated values for the ATR and ensure that the safety studies based on calculations are valid. Techniques include pile oscillator, noise, and pulsed neutron measurements.

4. APPROACH TO POWER

(1) Hydraulic Capability Tests -- The loaded ATR core will be subjected to normal full power coolant flow (reactor in the shutdown mode) for an extended time period followed by fuel removal and inspection before raising reactor power levels.

(2) Final Rod Performance Test -- In conjunction with the above hydraulic test, each control rod will be subjected to a final performance test.

(3) Stepwise Power Increase -- The ATR power level will be increased in steps with extended operation at each step. The following tests will be performed at each level:

(a) Measurement and evaluation of structural and internal temperatures

(b) Measurement and evaluation of radiation levels at specified locations

(c) Instrument surveillance including noise level

(d) Equipment performance

(e) Nuclear instrument calibration by either foil activation or thermodynamic measurements.

(4) Heat Transfer Tests -- Heat transfer tests to measure decay heat removal requirements will be made.

(5) Reactor Safety Stability Studies at Power -- The transfer function and power decay curves will be determined at one or more power levels and xenon studies will be made as the reactor power level is increased.

(6) Temperature Coefficient -- The temperature coefficient will be measured as a function of power during the approach to power period.

(7) High Power Flux Map -- Flux mapping will be performed with the reactor at high power.

(8) Fuel Inspection -- Prior to reaching full power, fuel will be unloaded and inspected for dimensional stability, corrosion, and metallurgical defects.

5. ATTAINMENT OF FULL POWER -- CORE I

The ATR will be operated at full power for one week with a balanced lobe power configuration. During this period, detailed data will be taken relative to reactor, process, and radiations systems to evaluate overall facility performance. Following shutdown, the core will be unloaded and subjected to detailed hot cell examination including destructive testing.

6. CORE II TESTS

Based on results of Core I performance, Core II tests will include some experiment loading and lobe power split at full power.

XIV-3. OPERATING PROCEDURES (PRELIMINARY)

The ATR operating procedures are presented in detail in the operating manual. Procedures presented below do not outline the extensive check list procedures required to be filled out and signed by the Shift Supervisor prior to reactor start-up. These check lists include valve arrangement for all systems (a secondary check is available at the console for critical block valves, such as the emergency coolant pump valves, by position lights), safety system condition, process system condition, and experiment systems including reactor safety system tie-in.

The following is a copy of the preliminary operating manual related to normal and abnormal operating procedures. These procedures were prepared prior to design completion. Updating of these procedures is in progress and the final draft of the operating manual will reflect the final design.

ATR OPERATING MANUAL

3.6 Normal Reactor Operation

3.6.1 Preoperational Check-out

1. All maintenance work is complete; the reactor vessel and the primary system are filled and prepared for operation.
2. The core is in place and the reactor head refueling ports and top-head shield are in place.
3. All reactor checkoff lists, including calibrations, have been completed and satisfied.
4. All reactor and auxiliary systems are checked out and in normal operation -- ready for reactor power operation.
5. Select one of the two motor-operated rheostats (MOR) with associated servo channel and regulating rod using the MOR select switch S-59 in preparation for automatic operation. By the same operation the alternate MOR rises to its maximum setting.
6. All safety, nuclear, and process circuit requirements must be satisfied in order to withdraw first the safety rods and then the shim control rods to bring the reactor to criticality and to power. At least one start-up channel test block and one Log-N test block and two level test blocks must be in place.

3.6.2 Reactor Start-up

A procedure panel mounted on the console for the convenience of the operator indicates by signal lights the stepwise completion of the start-up sequence.

3.6.2.1 Cold Core with Two Starting Channels Operating in Source Range

1. Turn keyed switch S-O to ON; and push the start button S-1.

2. The start-up warning horn is sounded automatically for ten seconds, and the circuits energize the REACTOR ON lights at the reactor console and throughout the building.

3. Adjust the selected MOR to the N_L position so that the reactor can be placed in automatic control at N_L .

4. Withdraw the fission chambers slightly and observe a drop in the count rate to assure that the neutron start-up level instruments are operating properly. Return the chambers to their original positions, maintaining on-scale readings.

5. Withdraw the safety rods to approximately three inches in a group and scram to show proper operation.

6. Withdraw safety rods in a group until all safety rods have reached their full-out limit switches. With all safety rods withdrawn the reactor should not be critical. If the reactor power level is less than $10^{-5}N_F$, an intermittent timer limits shim and safety rod motion to 0.2 second out of each one second.

7. The control circuitry will scram the reactor at SAG unless all safety rods are at their full withdrawn limit switches. The reactor will also scram if a safety rod is dropped during start-up and the power level is below SAG.

8. Manually position the alternate regulating rod to its full out position with manual regulating rod switch S-64; adjust the selected regulating rod to Range 4 of the regulator rod control.

9. Withdraw the outer shim control cylinders toward their rod positions to define the power split desired as determined in the ATRC. This must be done alternately in two groups (the north and east cylinders compose one group associated with the fission chamber in thimble T-2 and the south and west cylinders as the other group associated with the other fission chamber in thimble T-9) to prevent shadowing of both fission chambers simultaneously.

Note

A 30 second reactor period below $10^{-5}N_F$ will interrupt rod withdrawal. A 7 second reactor period above $10^{-5}N_F$ will interrupt rod withdrawal.

10. Withdraw the neck shims if desired. When the count rate recorder pen continues to increase after withdrawal has stopped, the reactor is critical.

Note

Steps 9 and 10 may be done concurrently.

11. At criticality, the count rate recorders (start-up level channels) will be several decades above the source level and the Log N recorders will start to show a power increase. As soon as the Log N recorders are reading on scale, withdraw the fission chambers, keeping them reading on scale.

12. Raise the power level above $10^{-5}N_F$. Above this power level, the outer shim rods may be positioned continuously. If the fission chambers are not fully withdrawn at SAG, they will automatically withdraw at SAG.

13. Push the run pushbutton after the reactor power level reaches SAG. Neck rod group withdrawal is not permitted in RUN regime. The reactor enters the RUN regime only if the fission chambers have been withdrawn.

14. As the reactor power level passes upward through SAG, a timer starts which runs for 11.5 seconds. At the end of this span of time, the reactor level channels' sensitivity changes by approximately two decades including the over-power trip setpoints. If the power level exceeds N_L as indicated by the Log N channels before the timer's cycle is complete, a scram will occur.

Note

The reactor will scram at $3N_L$ if the reactor is in the START regime.

15. Continue increasing the power level to N_L as indicated on the Log N channels until the reactor automatically goes into RUN AUXILIARY at N_L .

Caution

Not at any time shall more than one neck shim rod be in an intermediate position above $3N_L$ except during automatic power reductions.

16. The selected servo channel will maintain the reactor power at the level set on the active MOR (initially N_L).

17. Raise the MOR setpoint in steps to 20 percent full power. Should the regulating rod withdraw beyond its normal operating range (Range 5), it can be restored automatically to this range by manually withdrawing preferred shims (neck or outer shims). After the plant has stabilized, calibrate the N-16 and power level instrumentation based on ATRC rod positions and water power instrumentation. Check and adjust the lobe power ratios in proportion to the desired lobe power balance at N_F as computed by the data processing system.

Note

After reaching RUN, the operator may choose to go to power manually by using solely the shim rods. In RUN there is no manual control of the regulator rods and the MOR has no setpoint significance until the selection of RUN AUXILIARY. To prevent from automatically going into RUN AUXILIARY, either the regulating rod must be in Range 5 or 6 or the selected MOR must be off its lower limit when the power level reaches N_L .

18. Raise the MOR to other intermediate power levels and repeat procedure in Step 17.

19. Finally raise the MOR to N_F and establish the desired lobe power balance in accordance with the procedure in Step 17.

3.6.2.2 Cold Core with One Start-up Channel Operating in Source Range

1. Repeat steps 1 through 8 in 3.6.2.1.
2. Withdraw all neck shims in a single group until criticality is reached. This eliminates rod shadowing of the single start-up channel.

Note

A 30 second reactor period below $10^{-5}N_F$ will interrupt rod withdrawal. A 7 second reactor period above $10^{-5}N_F$ will interrupt rod withdrawal.

3. When the count rate recorder pen continues to increase after withdrawal has stopped, the reactor is critical. Increase power and maintain the count rate meters on scale by withdrawing fission chambers.

4. Withdraw neck shims in a group until power level reaches $10^{-5}N_F$ as indicated on Log N channels. Above $10^{-5}N_F$ the shim rods may be positioned individually.

5. By rod interchange between neck shims and outer shim control cylinders establish the predicted rod positions to define the power balance desired as determined in the ATRC. During these rod interchanges maintain power level between $10^{-5}N_F$ and $10^{-4}N_F$ as indicated on Log N channels.

6. Raise power level by withdrawing shims. If the fission chambers are not fully withdrawn at SAG, they will automatically withdraw at SAG.

7. Push the run pushbutton after the reactor power level reaches SAG. Neck rod group withdrawal is not permitted in RUN regime. The reactor enters the RUN regime only if the fission chambers have been withdrawn.

8. Repeat steps 14 through 19 in 3.6.2.1.

3.6.3 Full-Power Operation

1. After the reactor reaches full power at the beginning of a core cycle, the reactivity will go through a transient period while the water temperature, xenon poison in the reactor fuel and experiments, and other parameters come to equilibrium.

2. Under servo operation, full power is maintained by automatic movement of the active regulating rod and occasional manual shim withdrawal to keep the regulating rod within its normal operating range (Range 5).

3. When necessary the operator restores the lobe power balance by manipulation of the outer shim control cylinders and the neck shim rods. The lobe power balance is computed by the data processing system.

4. Numerous checks on the instrumentation and equipment are necessary during steady full-power operation to maintain continually the neutron environment for the tests being conducted.

3.6.4 Reactor Shutdown

1. After core burnup has proceeded to the point that the experiment requirements cannot be met because either the total reactor power or the lobe power balance cannot be maintained, the reactor is ready for shutdown and subsequent refueling. The operator has the option of a longer fuel cycle by a more balanced lobe power split. The full-power operating cycle is approximately 17 days.

2. Shutdown the reactor by pushing the manual scram pushbutton or by manually inserting of shim and safety rods.

3.7 Recovery Operations from Power Reductions

3.7.1 Recovery from Setback

1. A setback lowers the motor-operated rheostat (MOR) setpoint toward N_L . The reactor power follows the MOR demand by automatic insertion of the active regulator rod. This action continues until the cause is cleared or N_L is reached.

2. Investigate to determine the cause of the setback, and take action to remedy the cause.

3. Reset (lower) the reactor inlet temperature setpoint for flow control valve TCV-1-10 to the actual inlet temperature.

4. Return the MOR in steps to the original setpoint. If necessary, position the shims to restore the proper lobe power balance. Depending on the duration of the setback, the shim rods may have to be withdrawn temporarily farther than their previous positions in order to restore full power operation because of xenon present after the shutdown. As the transient xenon is burned up the shim rods will be inserted until equilibrium is restored.

5. After the reactor reaches full power, increase the reactor inlet temperature setpoint gradually to 125°F.

3.7.2 Recovery from Reverse

1. A reverse is a power reduction accomplished by the insertion of all neck shim rods, all the outer shim control cylinders, and the selected regulating rod at its manual operating speed of five inches per second. Reactor control reverts to manual. This action continues until the cause is cleared except the regulating rod continues in until it is full-in.

2. Investigate to determine the cause of the reverse, and take action to remedy the cause.

Note

Steps 3 through 10 apply if the reactor is critical or above after the reverse clears. Steps 11 through 22 apply if the reactor is subcritical after the reverse clears.

Partial Rundown

3. Perform Step 3 of 3.7.1.
4. Withdraw all partially inserted neck shim rods to their (full-out) previous positions.
5. Withdraw all outer shim cylinders to their previous positions.

Note

Steps 4 and 5 may be done concurrently.

6. Manually set the MOR and actuate the regulating rod manual-run auxiliary switch S-26 to return to RUN AUXILIARY.

7. Check the lobe power ratios and adjust shim rods to restore the lobe power ratios within the limits of 1.0 to 1.0 and 1.5 to 1.0.

8. If the power level is below $0.5N_F$, return the MOR to $0.5N_F$. Check and adjust the lobe power ratios and restore the ratios proportional to the desired balance at N_F . Return the MOR to N_F ; check and adjust the lobe power balance until N_F and the desired balance is achieved.

9. If the power level is above $0.5N_F$, return the MOR to N_F . Check and adjust the lobe power balance until N_F and the desired balance is achieved.

10. Perform Step 5 of 3.7.1.

Full Rundown

11. Withdraw all partially inserted neck shim rods plus estimated additional neck rods to compensate for xenon poisoning to their full-out positions.

12. Withdraw the outer shim control cylinders to their previous positions plus additional outer shims to complete compensation for xenon poisoning.

13. Lower the MOR at fast speed to N_L .

Note

Steps 11 and 12 may be done concurrently. Step 13 could also be done concurrently with 11 or 12 or with both.

14. Continue to withdraw outer shims and push the run pushbutton after the power level exceeds SAG; raise power level to N_L .

15. As the power level reaches N_L , observe that the reactor enters RUN AUXILIARY automatically.

16. Perform Step 3 of 3.7.1.

17. Check and adjust the lobe power ratios within the limits of 1.0 to 1.0 and 1.5 to 1.0.

18. Raise the MOR at fast speed to $0.5 N_F$.

19. Check the lobe power ratios and adjust the shims to restore the ratios proportional to the desired balance at N_F .

20. When the lobe power ratios have been achieved, increase the power level toward N_F in intermediate steps being careful not to exceed the lobe power ratios proportional to the desired balance at N_F at each step.

21. Maintain the power level at each intermediate step. Check and adjust shims to restore the desired lobe power ratios at each step. Continue increasing the power level toward N_F . As xenon is destroyed and excess reactivity becomes available, raise the MOR toward N_F . Continue this procedure until the power level is N_F and the proper lobe power balance is achieved.

22. Perform Step 5 of 3.7.1.

3.7.3 Recovery from Fast Recovery Scram

1. A fast recovery scram is a quick power reduction accomplished by the release of the fast recovery scram safety rod. A full scram will occur in 60 seconds if the fast recovery scram signal is not cleared and/or the scrambled rod has not been withdrawn off its lower limit switch. (See 3.7.4 for this case.)

2. Investigate to determine the cause of the fast recovery scram and take action to remedy the cause.

3. Immediately lower the MOR at its fast speed to $0.5 N_F$ (do not raise MOR until safety rod is fully withdrawn -- Step 5).

4. Perform Step 3 of 3.7.1.

5. Withdraw scrambled safety rod to its full-out position.

6. In the event that the reactor goes critical during the safety rod withdrawal, the power level will increase to N_L where reactor control system will automatically enter RUN AUXILIARY and, subsequently, the power level will be maintained at N_L until safety rod is fully withdrawn.

7. Withdraw outer shim control cylinders to complete compensation for xenon poisoning.

8. Repeat steps 11 through 22 in 3.7.2.

3.8 Special Operations

3.8.1 Low-Power Operation with Open Head

3.8.1.1 General

The reactor is operated at low power with the primary system at atmospheric pressure. This operation is conducted with one or more refueling ports removed and a clean core installed in the vessel. This operation is conducted at N_L with the reactor under automatic control for flux mapping.

3.8.1.2 Preoperational Check-out

1. Steps 1 through 6 in 3.6.1 must be satisfied.
2. One or more reactor refueling ports must be removed.
3. Check that both emergency coolant pumps are operating.
4. The UCW system must be set up and in operation for heat removal on the secondary side of the primary heat exchanger.
5. The required flux wires are inserted in their specified locations.

3.8.1.3 Reactor Start-up with Two Start-up Channels

1. Repeat Steps 1 through 14 in 3.6.2.1.
2. Insert the four linear power level detectors located in thimbles T1, T5, T8, and T12 until their recorders read N_F while all other power level instrumentation reads N_L . This provides power level reduction trip setpoints and an electronic scram from the level channels two decades below their normal settings.

3.8.1.4 Reactor Shutdown

1. After flux mapping operations are complete the reactor is shutdown.
2. Shut down the reactor by pushing the manual scram push button or by manual insertion of shim and safety rods.

3.8.2 Operation with Equipment Failure

3.8.2.1 Safety and Shim Rod Limit Switches

1. Each safety and shim rod position is indicated on the reactor console (full scale indication and upper and lower indicating lights) and is also available to the operator through the data processing system in print out

form. One safety rod upper limit switch may be bypassed through a test block located on the reactor panel in the reactor control room. The use of these test blocks is necessary when failures of these limit switches occur and the actuation of the failed switch is required to provide permissives for other operations. These test blocks are under strict administrative control. Their use is only for checking, special tests, and maintenance purposes and when the other indications demonstrate the true position of the rod.

3.8.2.2 Servo Channel Failure

1. If possible, withdraw the active regulating rod to its upper limit.

2. Actuate MOR channel selector switch S-59 to the alternate channel. In the event that the alternate regulating rod is not fully withdrawn at the time of this switchover, the reactor will scram.

3.9 Abnormal Procedures

3.9.1 Fuel Element Failure

1. The reactor safety system will shut down the reactor due to high gamma activity in the primary coolant.

2. Remove failed fuel from core.

3. Refuel reactor.

4. Start up reactor in accordance with 3.6.2.

3.9.2 Experiment Loop Rupture

The reactor safety system automatically shuts down the reactor due to one or more of the following conditions:

(a) High neutron flux level

(b) Short reactor period

(c) Low coolant pressure

(d) Low coolant flow

(e) High specimen temperature

(f) Fission break

2. Remove specimen and failed pressure tube. (See Loop Operating Manual).

3. Insert new pressure tube (See Loop Operating Manual).

4. Refuel reactor in accordance with 3.10.3.

5. Start up reactor in accordance with 3.6.2.

3.9.3 Instrument and Control Bus Power Failure

1. The reactor safety system scrams the reactor due to low voltage to the magnet amplifiers.
2. Restore bus power supply.
3. Start up the reactor in accordance with 3.6.2.

3.10 Handling Procedures

The handling tools have been given earlier.

3.10.1 Precautions

1. All handling procedures described below should be conducted under the strict surveillance of a health physicist. Airborne contamination and radiation level should be checked continually.
2. The reactor vessel water level should never be dropped below a 100-foot elevation except when disconnecting experiment leads during instrument capsule handling when the level is dropped to the 93-foot elevation.
3. Care should be exercised in the prevention of foreign materials from falling into the reactor vessel.
4. Spillage of primary (reactor vessel) water onto the movable top shield and working panels and in contact with operating personnel should be avoided.

3.10.2 Reactor Preparation

1. The reactor should be shut down.
2. Depressurize the primary coolant system and flush to reduce the primary water activity.
3. The emergency coolant pumps should be operating and providing continuous primary coolant flow during these operations.
4. Verify that the top-head area ventilation system is in operation.
5. The drop tube cover is in place.
6. Tag out the operation of the primary pumps to prevent accidental starting when primary coolant system is depressurized and the re-fueling ports are removed.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

XV. ADMINISTRATIVE AND PROCEDURAL SAFEGUARDS

XV-1 Organization and Function

XV-2 Administrative Controls and Procedures

XV-3 Review and Authorization of Tests, Experiments, Modifications, and Safety Status

XV-4 Training

XV-5 Summary

**THIS PAGE
WAS INTENTIONALLY
LEFT BLANK**

XV-1. ORGANIZATION AND FUNCTION

1. PHILLIPS PETROLEUM COMPANY AED ORGANIZATION STRUCTURE

The Phillips Petroleum Company was selected as operating contractor for the Advanced Test Reactor by the United States Atomic Energy Commission. Specific responsibility to operate the ATR rests with the Atomic Energy Division (AED) of Phillips' Research and Development Department located in Idaho Falls, Idaho.

The AED, at present, has responsibility for operating the MTR, ETR, ICPP, conducting the Special Power Excursion Reactor Test (Spert), Safety Test Engineering Program (STEP), and SNAP Transient Studies (SNAPTRAN). Four low power reactor systems are also operated by Phillips AED. These are ARMF-1 and -2, ETRC, and ATRC. In addition to the above operational and testing responsibilities, Phillips AED carries out extensive research and development programs in the fields of nuclear physics and chemistry, reactor physics, applied mathematics, instrumentation, metallurgy, reactor engineering, design, and chemical processing technology.

The organization structure of Phillips Petroleum Company AED is shown in Figure XV-1-1. Management of the AED is under the overall direction of a Division Manager, responsible to the Vice President, Research and Development Department, for prosecution of Phillips' contractual obligations to the Atomic Energy Commission. The AED has five major subdivisions: Operations, Nuclear and Chemical Technology, Nuclear Safety Technology, Engineering, and Administration, each headed by an Assistant Manager accountable to the Division Manager for all activities under his direction.

2. ATR OPERATIONS STRUCTURE

Operation of the ATR is the responsibility of the ATR Plant Superintendent who is accountable to the TRA Plant Manager. The reactor operations organization is shown in Figure XV-1-2.

The function of the ATR Operations Branch is to operate the reactor, supporting facilities, and experiments within a framework of established safe operating practices, administrative procedures, and experimental requirements.

The Operations Superintendent administers company policy as delegated to him and enforces at the branch level the operating policies and procedures established by management. The Superintendent consults when necessary with other organizations such as Reactor Engineering, Reactor Safeguards Committee, Plant Engineering, etc, on problems which require technical or administrative assistance. The Assistant Superintendent is sufficiently familiar with the Superintendent's duties to be able to act for the superintendent in his absence. The Assistant Superintendent's duties include: (a) following all operations to assure that the prescribed operating conditions are maintained; (b) troubleshoots special problems; (c) maintains liaison with other support organizations including maintenance; and (d) reviews and evaluates operating records and log books.

PHILLIPS PETROLEUM COMPANY

ATOMIC ENERGY DIVISION

IDAHO FALLS, IDAHO

JUNE 1, 1964

FIG. XV-1-1 PHILLIPS PETROLEUM COMPANY AED ORGANIZATION STRUCTURE.

FIG. XV-1-2 ATR OPERATIONS ORGANIZATION.

A coordinator reports directly to the Assistant Superintendent. The functions of the Coordinator are: (a) to maintain liaison between the Operations Branch and Maintenance Service Branch; (b) to review proposed experiments scheduled for insertion in the reactor for adequacy of design and operational feasibility; (c) to schedule maintenance for operating equipment; (d) to review step-by-step procedures for inserting and removing experiments from the reactor; and (e) to maintain close surveillance over the operating conditions of all reactor experiments and to trouble-shoot experiment problems.

The Shift Supervisors report to the Assistant Superintendent and direct the work of the shift engineers, operators, and technicians engaged in around-the-clock operation of the reactor and supporting facilities. In their capacity, the shift supervisors are responsible for the entire reactor and plant on off shifts, weekends, and holidays. One supervisor, in addition to being a relief shift

supervisor, has responsibilities for the preparation and updating of operating manuals and for the training of personnel.

The Canal and Utility supervisor is responsible for all work performed in the canal including shipments, fuel cutting operations, and experimental handling. He is responsible for maintaining detailed records and logs of canal inventory.

A Senior Reactor Engineer and several Reactor Engineers are assigned on each shift to operate the reactor console and loop experiment consoles. The Senior Reactor Engineer acts as the chief operator, making operating adjustments as required and logging data as required by operating procedures. He conducts routine checks on reactor experiments and directs and assists with changing reactor fuel and experiments when the reactor is shut down.

Each shift is also staffed with a sufficient number of operators and operator helpers to operate process equipment and auxiliary systems.

In order to maintain close surveillance over the reactor control system and nuclear instrumentation, the Operations organization contains a Reactor Instrument Section consisting of a Reactor Instrument Supervisor (engineer) and several Reactor Instrument Technicians. It is the responsibility of the Instrument Supervisor, assisted by the technicians, to maintain, check, and repair all reactor control and nuclear instrumentation and assure proper functioning of reactor instrumentation at all times. The Reactor Instrument Supervisor is also responsible for training the Instrument Technicians and for assignment of their preventative maintenance duties. One Reactor Instrument Technician is assigned to each shift, reporting to the Shift Supervisor on off shifts, weekends, and holidays.

3. SUPPORT ORGANIZATIONS

Several Branches and sections of the AED provide direct technical support to the ATR Operations Branch. A list of these organizations and functions follows:

3.1 Reactor Engineering Branch

This branch operates the ATRC providing detailed flux and power profiles for each core loading, examines experiment effects, reactivity, control rod worths, expected cycle life, and uniformity of fuel and burnable poisons. In addition, the branch provides metallurgical services, and hydraulic and heat transfer assistance including problems relating to safety.

3.2 Mathematical Analysis and Machine Computations Branch

This branch provides reactor physics calculations, mathematical assistance, problem coding, and machine computations. In addition, special programs for the ATR data information and computation equipment will be developed by this branch.

3.3 Instrument Development and Technical Services Branch

The Instrument Development and Technical Services Branch consists of specialists in detectors of all types, data system, and electronic systems. The services of this branch are available in support of operation of the ATR. In addition to systems and equipment specialists, the Branch performs detailed nuclear and process kinetic studies and has complete analog and digital computing equipment available for studies.

3.4 Division Engineering

Engineering services in support of ATR operations are provided by Division Engineering. This organization includes the following branches:

- (1) Design-Construction Liaison
- (2) Design Engineering
- (3) Plant Engineering.

The Design-Construction Liaison Branch provides engineering liaison and coordinates design and construction activities with AEC on all construction jobs in which the AED is involved. This service has been supplied for the ATR since the initiation of design.

The Design Engineering Branch is responsible for supplying the ATR Operations Branch with mechanical design engineering and drafting service, shop liaison, cost estimating, mechanical engineering consultation, etc. All proposed structural or mechanical changes, additions, or modifications to the reactor and supporting facilities are designed and approved by this branch.

The Plant Engineering Branch is responsible for providing process engineering assistance to the ATR Operations Branch. The responsibilities include engineering assistance with process problems concerning hydraulics, heat transfer, corrosion, water treatment, waste disposal, process equipment operational problems, etc. In addition, the responsibility for mechanical and hydraulic testing of reactor fuel and other in-pile components is assigned to this branch.

4. ATR START-UP COMMITTEE

An ATR Start-Up Committee has been established by the Division Manager to direct and coordinate all activities associated with the start-up of the ATR. This committee is composed of senior AED staff members and includes the ATR Plant Superintendent, as chairman, Assistant Manager-Nuclear and Chemical Technology, Plant Engineering Branch Manager, Design-Construction Liaison Branch Manager, Design Engineering Branch Manager, Mechanical Services Branch Manager, TRA Health and Safety Director, and Assistant Manager Project Engineering.

This committee is responsible for reviewing the results of all component continuity (CC) tests, and systems operations (SO) tests to ensure compliance with design specifications and requirements and initiate modifications when necessary.

The committee is responsible for planning and execution of all pre-neutron and post-neutron tests including approach and attainment of full operating power of the ATR. Specific responsibility for all nuclear and research and development tests during the start-up is assigned to the Assistant Manager-Nuclear and Chemical Technology.

The committee is also responsible for seeing that all phases of the start-up program having nuclear safety implications are reviewed and approved by the appropriate safety committee.

5. REACTOR SAFEGUARD COMMITTEE

The safe operation of a test reactor and associated experimental programs requires a continuing evaluation of the hazards and risks posed by the changing nature of this type of operation.

The existing MTR-ETR Safeguard Committee has successfully performed this evaluation function for the MTR and ETR during the major portion of the reactors' operating histories. This committee will be extended to include the ATR.

The committee is composed of senior members of the Phillips' organization appointed by the Division Manager. The members include a chairman (full time), Senior Member of Plant Engineering, Senior Physicist, Senior Health Physicist, and Senior ATR Operations supervisor.

The responsibilities of the committee are: (a) to evaluate the hazards attendant to various ATR operations and experiments and to establish and maintain acceptable risk levels within approved operating limits, and (b) to review and prevent the recurrence of incidents which directly, or indirectly, involve radiation hazards at ATR.

The following are activities that require Reactor Safeguards Committee Approval:

- (1) Insertion of all experiments and experimental equipment including operating conditions and safety circuit setpoints. Modifications to experimental equipment or changes in safety circuit setpoints,
- (2) Any physical change in the reactor, change in reactor control circuitry or safety circuit set points,
- (3) Any change in reactor or experiment operating procedure or method which might conceivably affect reactor or experiment safety,
- (4) Continued operation of the reactor or an experiment after deviation from approved operating conditions which might conceivably affect the safety of the reactor or experiment, and
- (5) All changes in the ATR core loading which may affect the safety of the experiments or the reactor.

Approval for activities falling in the above areas is mandatory. It is the responsibility of Phillips' line supervision to determine when their actions require Reactor Safeguard Committee approval and to ensure that operations performed under their supervision conform to existing Reactor Safeguard Committee approvals.

6. SAFEGUARD REVIEW COMMITTEE

A Safeguard Review Committee, appointed by the Division Manager, is responsible for reviewing once per year the policies and procedures used by all operations organizations. The reviews are conducted to determine that current operating instructions and practices are sufficient to ensure adequate margins exist with respect to nuclear and radiation safety.

7. NUCLEAR SAFETY COMMITTEE

The Nuclear Safety Committee, appointed by the Division Manager, consists of five senior AED staff members with duties and responsibilities as follows:

- (1) Search out and initiate investigations on any activities within the division having the potentiality of a criticality accident which, in the opinion of the committee, may not be adequately safeguarded under present procedures.
- (2) Make recommendations to responsible supervisors on ways and means of remedying any deficiencies that may be observed in the safeguarding of potential criticality areas. Copies of recommendations to be sent to AED Manager.
- (3) Conduct investigations as requested in writing by responsible supervisors to determine nuclear safety of various operations.

A review of all storage and shipments of fissionable materials is made if there is any possibility of criticality.

8. HEALTH AND SAFETY

The Health and Safety Branch is part of the Operations organization. The responsibility for operational safety resides in the line organization, with Health and Safety personnel in an assisting role. However, the Health and Safety organization also performs an inspection role independently of the operating organization and is vested by the Division Manager with the authority and responsibility to shut down any operation which it considers unsafe.

Health Physics personnel will be provided at ATR around-the-clock on a shift basis. These personnel are under the direction of a Health Physics supervisor. The group has the responsibility for providing the following services:

- (1) Radioactivity surveys
- (2) Routine area monitoring
- (3) Radioactive shipments monitoring
- (4) Monitoring and limiting the release of radioactive waste of all types
- (5) Routine first aid
- (6) Assistance in training personnel.

Work performed in any area which is potentially a radiation hazard must receive prior survey and approval of the Health Physics group, as specified by safe work permit procedures.

The AED Safety Engineer will assign one man to the ATR specifically trained in industrial safety. It is the responsibility of this individual to provide industrial safety service by reviewing all safe work permits, providing special safety equipment, and instructing personnel in safety practices, fire prevention, and use of safety equipment.

9. ACCIDENT REVIEW COMMITTEE

The Accident Review Committee, appointed by the Division Manager, is composed of senior AED staff members. The function of this committee is to review all incidents involving injury (including overexposures) and/or property damage and investigate and prepare a formal report when necessary. The report will include a complete description of the incident and recommendations for corrective action. The responsible Assistant Manager will review the recommendations for corrective action and notify the Accident Review Committee and the Division Manager of corrective action taken.

10. PROJECT ENGINEERING

The Project Engineering Branch supervises the experimental programs associated with the test reactors. Two major functions of this branch are to review experiment designs to ensure that adequate safety margins are available and that all factors influencing the safety of personnel and equipment have been considered, including hazards analysis, and provide technical assistance to experimenters in the following areas:

- (1) Ensure design compliance with requirements for inserting experiments in the test reactors

- (2) Operational feasibility
- (3) Provide experimenters with data on reactor performance and arrangement to aid in design
- (4) Present a safety review of each experiment to the Reactor Safeguards Committee including safety circuit setpoints
- (5) Supervise installation, inspection, preinsertion, testing, operational training, and removal of experiments
- (6) Prepare detailed operating instruction for each experiment and maintain surveillance of experiment behavior
- (7) Supervise all experiment maintenance requirements.

11. TEST REACTOR OPERATIONS TRAINING AND QUALIFICATIONS COMMITTEE

This committee has the responsibility of ensuring that all operating personnel are qualified. The committee is also responsible for preparing and updating the manuals and content of the formal training course "Fundamentals in the Operation of Nuclear Test Reactors".

The committee consists of a chairman and four senior AED staff members from the various organizations.

XV-2. ADMINISTRATIVE CONTROLS AND PROCEDURES

The complexity of test reactor operation with associated experiments requires a system of administrative control and procedures more stringent than is usually practiced in power reactor operation. The reasons stem from the added hazards of the experiments, the high power densities required to produce the intense neutron flux needed by the experiments, and the complexity of maintenance operations during frequent shutdown periods.

Operation of the MTR and ETR has resulted in the development of practices and procedures leading to safe and efficient reactor performance.

1. STANDARD PRACTICES MANUAL

The Atomic Energy Division has established a set of standard practices for each reactor facility establishing uniform rules that must be complied with in the interest of safety and efficient operation. These rules are contained in a standard practices manual and include specific instructions and operational criteria which are mandatory. Examples of some of the mandatory rules in the standard practices manual are: required operational safety circuits for reactor start-up, method of removing and inserting control elements and fuel, recovery from a power reduction, canal storage and handling requirements, instrumentation required during shutdowns, removal of experiments, and supervisory responsibilities.

A standard practices manual will be prepared for the ATR and approved by the Nuclear and Chemical Technology Assistant Manager and the Chairman of the Reactor Safeguards Committee before loading fuel into the ATR. The manual is available to all operating personnel, and every man in the Operations Branch is responsible for becoming thoroughly familiar with the instructions and rules contained in the manual pertaining to his job. The manual is subject to an annual review by the Safeguard Review Committee.

2. OPERATING MANUALS

The ATR operating manual has been prepared and is available to operating personnel for training purposes, operational testing of components and systems and to up-date as required prior to operating the reactor. The manual contains system descriptions, step-by-step operating procedures, and other information pertinent to the equipment described.

3. MAINTENANCE CONTROL

The extensive maintenance work necessary to support test reactor operations requires procedures that define responsibility. The standard practices manual provides this definition. The shift supervisor is responsible for all maintenance

work in the reactor tank and this work is performed under operational supervision. In addition, all maintenance work on equipment that relates to plant safety is under operational supervision.

XV-3. REVIEW AND AUTHORIZATION OF TESTS, EXPERIMENTS, MODIFICATIONS, AND SAFETY STATUS

1. PROCEDURES AND AUTHORIZATION FOR PERFORMING REACTOR TESTS

The procedures and authorization to perform tests or change operating conditions that fall outside of normal operations are as follows:

- (1) A written report of the proposed test or operational change is prepared including detailed procedures
- (2) A review and approval of the Nuclear and Chemical Technology Assistant Manager is required
- (3) A review and approval of the Reactor Safeguards Committee is required
- (4) No one outside the Operations organization is permitted to do any work affecting the reactors and the associated facilities without the approval of the responsible Operations supervisor.

2. PROCEDURES AND AUTHORIZATION FOR INSERTING EXPERIMENTS

The following procedures and authorizations are required for all experiments to be inserted in the ATR:

- (1) Technical assistance during experiment design phase by the Project Engineering Branch.
- (2) Review of the final design by the Project Engineering Branch.
- (3) Presentation of the experiment design, including setpoint list actuating reactor safety circuits to the reactor Safeguards Committee. (A written presentation of the experiment precedes an oral presentation, both prepared by the Project Engineering Branch.)
- (4) Formal approval is given for experiment insertion by the Reactor Safeguards Committee if satisfied that the experiment meets all requirements to ensure the safety of personnel and equipment.

3. PROCEDURES AND AUTHORIZATION FOR MODIFYING FACILITY

Any modifications to the reactor, including control and safety circuitry, setpoints, and core loading, must first be approved by the Nuclear and Chemical Technology Assistant Manager and the Reactor Safeguards Committee.

4. REVIEW AND SURVEILLANCE OF REACTOR SAFETY QUESTIONS

A Reactor Surveillance Section, reporting directly to the Assistant Manager of Nuclear and Chemical Technology, has a continuing responsibility for exercising technical surveillance over all test reactors to ensure operations are kept within specified technical limits and standards of operational behavior as regards reactivity, control parameters, reactor circuitry, stability, etc. This surveillance is exercised by both periodic checks of the reactors and trouble shooting when unusual difficulties are encountered during operation.

5. ATOMIC ENERGY COMMISSION AUTHORIZATION

In addition to the approvals and authorizations stated in the previous sections, some reactor modifications and operating parameter changes require Atomic Energy Commission approval. These areas are clearly set forth in the standard practices manuals.

XV-4. TRAINING

Phillips Petroleum Company, Atomic Energy Division, has developed and operates a training program (Fundamentals in the Operation of Nuclear Test Reactors) in addition to specific training programs related to each facility providing the necessary operating knowledge. Training programs in fire prevention, fire fighting, industrial safety, and radiation are also provided.

1. OPERATIONAL TRAINING

The training program for reactor operation personnel adopted by the Phillips Petroleum Company, AED, consists of formal classwork in "Fundamentals in the Operation of Nuclear Test Reactors" (FONTR) and practical training at his assigned location.

1.1 Formal Training

Formal training requires completion of the FONTR course by all ATR reactor engineers and technicians. The course requires completing assignments and written examinations in addition to classwork in the following areas:

- (1) History, Organization, Health and Safety, and Regulations
- (2) Reactor Science and Technology
- (3) MTR Reactor Design and Operation
- (4) ETR Reactor Design and Operation
- (5) Engineering and Design Considerations for In-Pile Tests
- (6) ATR Reactor Design and Operation (in preparation).

The classes are presented by a senior staff member of the AED (full time) assisted by staff specialists in the various areas covered by the course.

1.2 Practical Training

All ATR operating personnel will receive practical training at the ATR before operating the reactor. Performing the various systems operation tests, check-out of equipment, preparing and updating various manuals provide practical training and familiarization for ATR operating personnel in addition to more formal lectures by senior ATR operating supervisors.

Additional low power operational training will be obtained by using the ATR Critical Facility for this purpose.

Practical training at the ATR is a continuing process in which each shift supervisor functions as an instructor. In this manner, all operating personnel are kept up-to-date with any new developments.

1.3 Qualifications

The Operations organization uses a four point qualification program for all operating personnel. This program, along with educational and experience requirements, applies to all shift supervisors, reactor engineers, and technicians. The program consists of the following:

- (1) Successful completing of the FONTR course
- (2) Practical training (total time required is approximately one year)
- (3) Formal endorsement by the ATR superintendent
- (4) Oral examination by the Test Reactor Operations Training and Qualifications Committee.

The educational and experience requirements of the operating personnel are as follows:

- (1) Shift Supervisor -- An experienced engineer with several years of formal and on-the-job training, familiar with the elements of nuclear theory.
- (2) Reactor Engineer -- A salaried employee with a B. S. degree in Engineering, or one of the physical sciences, or equivalent experience in a refinery, steam, or chemical plant. Must be familiar with the basic elements of nuclear theory.
- (3) Technician -- A salaried employee with a B. S. degree in Engineering, or one of the physical sciences, or equivalent experience in a refinery, steam, or chemical plant.

2. FIRE-SAFETY-RADIATION TRAINING

All Phillips AED personnel are required to attend lectures in the fields of fire prevention, safety, and radiation protection given by specialists in the respective fields. Training is a continuous process and the results of these courses are reflected in the excellent safety records accumulated over the years by Phillips Petroleum Company AED in operating various facilities at the NRTS.

Special fire fighting courses are given to personnel working in areas that might require this knowledge.

Health-Physics procedures are stressed continuously throughout each facility in addition to formal lectures.

Safety lectures augmented by continual supervisory training are given to help maintain a safety orientated attitude among all AED personnel.

XV-5. SUMMARY

Phillips Petroleum Company, Atomic Energy Division, has developed administrative and procedural safeguards that have resulted in many years of safe, efficient operation of test reactors. Separation of operating, safeguard review, and surveillance functions into independent organizations provides a reliable system to ensure safe reactor operation.

The support organizations provide technical assistance in areas where unexpected problems occur. Control of all reactor and supporting facility operations rests with a line organization with definite responsibilities and chain of authority.

Operating personnel are required to meet high standards of education, experience, training, and character prior to examination by a qualifications committee.

The safe operation of reactors including conservative design, safety circuitry, various reviews, and procedures, still ultimately depends on an operating attitude that considers safe operation of paramount importance. Past performance with the MTR and ETR has demonstrated that Phillips Petroleum Company, AED, places much emphasis in promoting safety attitudes and safe practices in test reactor operation.

XVI. ACCIDENT ANALYSIS

XVI-1 Reactivity Additions

XVI-2 Nonnuclear Accidents

XVI-3 Reactor Damage Evaluation

XVI-4 References

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

XVI-1. REACTIVITY ADDITIONS

1. GENERAL

The evaluation of the consequences of a nuclear excursion in the ATR, the potential sources of reactivity additions, their magnitude, and rates of addition are analyzed in this chapter.

The analyses are based on experimental data obtained in the ATRCE and ATRC. The research and development programs to investigate ATR fuel element heat transfer and burnout have not been completed, and the analyses presented in this section are, therefore, based on results to date. Beginning- and end-of-life core parameters are based on the results of calculations previously described. Scram setpoints used in the calculations were 115 percent N_F whereas proposed setpoints (1.2 N_F) permit adequate sequential corrective action (setback) prior to a scram.

2. ACCIDENTAL REACTIVITY ADDITION FROM CONTROL ELEMENTS

2.1 Analysis of Possible Causes

The ATR reactor has four types of control and safety elements with associated drives. Each of these control and safety mechanisms is gear driven to provide positive mechanical motion. Ganged operation of all shim and safety rods is prohibited by interlocks. The reactor is subcritical at beginning of core life with all safety rods removed. Hence, an electrical or mechanical malfunction in the safety rod drive mechanism cannot cause a nuclear excursion. In addition, each loading variation is checked experimentally in the ATRC to ensure adequate shutdown margin is available in the ATR.

Ganged operation of all shim rods is permitted for one of every five seconds after the safety rods are at their upper limit only until the reactor reaches $10^{-5}N_F$. The maximum reactivity addition rate for simultaneous operation of all shim elements (neck and outer) is 0.058 percent $\Delta\rho$ per second if the above timer and neck shim interlock fail.

The most severe malfunction of the neck shim drives is an accidental continual operation which would drive the neck shims to their outer position. This would provide a maximum reactivity addition of 0.02 percent $\Delta\rho$ per second.

The outer shim control cylinders are driven in pairs by worm and wheel-type gear reduction systems mounted above the cylinders. Mechanical stops in the reflector and in the gears prevent cylinder over-rotation. Three types of failures are possible.

- (1) An electrical malfunction could withdraw the outer shim cylinders at their maximum rate corresponding to 0.026 percent $\Delta\rho$ per second when all 16 cylinders are moving simultaneously.

- (2) A failure of the drive line system is possible between the motor and the gear reduction system. This system has self-locking gears which stop cylinder motion.

(3) A failure of the shafting could occur between the gears and the cylinder, but the friction of the control cylinder bearing would prevent rotation of the cylinder. In addition, the cylinders have been designed to use coolant flow to exert a torque in the direction of negative reactivity.

Therefore, even with failures in all the shim rod drive mechanisms (both neck and outer cylinders), the reactor, with the safety rods withdrawn, has a maximum reactivity addition rate of less than 0.06 percent $\Delta\rho$ per second.

A higher ramp reactivity addition rate could result from a regulator rod failure in the power range. The resulting ramp rate is 0.6 percent $\Delta\rho$ /second, but only until the total rod worth (0.2 - 0.4 percent $\Delta\rho$) is inserted.

In summation, the maximum reactivity addition rates from the control system are limited to less than 0.1 percent $\Delta\rho$ per second for power levels less than 1 percent of full power with all shim and safety rods in ganged operation or in manual control above 1 percent of full power. Above 1 percent of full power, the maximum operational reactivity addition rate is approximately 0.6 percent $\Delta\rho$ per second in automatic control, but it is limited in magnitude to a total of 0.3 percent $\Delta\rho$.

2.2 Start-Up Accident

Reactor start-up accidents may result from an equipment malfunction or an operator error. At least one start-up channel must be reading on scale before any control rods may be moved. From the source level to 10^{-5} of full power, rod motion is limited to intermittent movement by an automatic timing device that limits average reactivity addition to 1/5 the maximum available from a continuous withdrawal. Timer failure prevents rod withdrawal. Therefore, at power levels below 10^{-5} of full power, the reactor is protected by interlocks, the intermittent timing device, and the period monitoring circuit. To have a rod withdrawal accident at the maximum combined rate of all shims and safety rods (0.1 percent $\Delta\rho$ /second) below 10^{-5} of full power would require two equipment failures and an operator error. In addition, the reactor is protected by a low-level trip circuit which can scram the reactor at 1 percent of full power if the elapsed time from 0.1 to 1 percent full power is too short (less than 11 seconds). This low-level trip circuit provides an independent backup of the period circuitry. Therefore, between source level and 1 percent of full power, several levels of safety action exist to prevent an excursion due to control rod withdrawal.

2.21 Normal Operation. The reactor power response to continuous reactivity addition rates has been evaluated by analog techniques to simulate a start-up accident with normal flow and pressure. The calculations incorporate only the flux level scram safety action and the effects of the negative temperature coefficient. For purposes of analysis, the reactor is assumed to be initially critical 5 decades below full power (the lowest level from which continuous rod withdrawal is permitted). The reactivity ramp rates are varied up to 0.2 percent per second, which is two times the maximum rate with simultaneous motion of all safety and shim rods in the core. The flux level scram is the only safety action assumed to occur, and a setpoint of 115 percent of full power was used. The actual scram setpoint during a start-up would be

1 percent of full power, but the higher value was used to demonstrate the inherent safety of the reactor during an excursion.

Figure XVI-1-1 shows the peak power as a function of the moderator coefficient for a 0.1 percent per second start-up accident initiated five decades below full power. Both beginning- and end-of-core-life conditions are presented. The results show that for the nominal moderator coefficient ($-1.37 \times 10^{-4} / ^\circ\text{F}$ for beginning of core life and $-0.97 \times 10^{-4} / ^\circ\text{F}$ for end of core life), the start-up accidents peak at 115 and 116 percent full power, respectively. The burnout value at the beginning of core life is 125 percent of full power. For a start-up accident with twice the maximum rod withdrawal rate, 0.2 percent per second, initiated 10 decades below full power, the peak power reached for both beginning- and end-of-core-life nominal moderator coefficients is less than 117.5 percent full power. Beginning-of-core-life conditions permit moderator coefficients as low as $-0.7 \times 10^{-4} / ^\circ\text{F}$ before the maximum rod withdrawal rate start-up accident at 10^{-10} full power causes the peak power to reach 119 percent full power. Increasing the scram setpoint to $1.2 N_F$ increases the values shown in Figure XVI-1-1 from 115 and 116 percent to 120 and 121 percent, respectively. An ample margin exists between the peak powers of these accidents and the beginning-of-core-life burnout level; therefore, these accidents present no problem to safe operation of the ATR.

FIG. XVI-1-1 START-UP ACCIDENT AT 10^{-5} FULL POWER WITH THE MAXIMUM ROD WITHDRAWAL RATE, 0.1 PERCENT $\Delta\rho$ PER SECOND.

The maximum reactivity rate assumes that the safety rods are withdrawn simultaneously with the shim elements. A check was made to see if adequate plant protection exists for a start-up accident occurring several decades below full power such that the reactor scrams when the safety rods were only 12 inches withdrawn (21 inches above the bottom of the active fuel). It was found that when the rods are withdrawn 12 inches, the reactivity removed in the first 0.042 seconds is identical to that removed for the fully withdrawn case, 36 inches. Furthermore, before the first 0.118 second, the rods will remove more $\Delta\rho$ when only 12 inches initially withdrawn. This is due to the fact that for the partially withdrawn case, the rods are on the steepest portion of the S-shaped rod worth curve.

2.22 Open-Head Operation. A special operating procedure for the ATR at low power to conduct flux mapping at reduced flow and open fuel handling ports on the vessel has been specified. During open-head operation, the reactor flow is reduced to 5400 gpm, and the maximum operating power level is reduced to 2.5 MW (N_L). These changes in the mode of operation affect many of the parameters that define the kinetic characteristics of the reactor. In addition to the thermal parameters, the safety rod characteristics also change, because the accelerating force decreases when the flow decreases.

The criteria for evaluation of the start-up accident during low-flow operation is that the reactor power should not exceed the burnout power.

A burnout power level of 6.6 MW was calculated for beginning-of-core-life conditions yielding the lowest value (during cycle) since flux peaking was at a maximum. The safety rod scram acceleration was reduced to 1.5 g.

Figure XVI-1-2 shows the peak power of start-up accidents at the maximum rod withdrawal rate, 0.1 percent $\Delta\rho$ /second, as a function of the scram setpoint at an initial power level of $10^{-3} N_L$ for beginning- and end-of-core-life nominal conditions. The lowest level from which continuous rod withdrawal is permitted is $10^{-3} N_L$ ($10^{-5} N_F$), and the curves representing this accident lie below the beginning-of-core-life burnout level. The end-of-core-life burnout level is higher than at the beginning-of-core-life because flux peaking is reduced.

2.3 Rod Withdrawal Accident at Power

The reactor is protected from excursion in the power range by period scrams and power level scrams. A period of one second or less or overpower will scram the reactor. Also, a lowering of the automatic control setpoint (setback) will occur when the reactor power exceeds certain preset values between full power and the scram level. Thus, both corrective and safety actions are available to prevent excessive excursions in the power range.

The response of the reactor, while operating in the power range, to ramp reactivity additions was calculated by using an analog simulation of the reactor including reactivity temperature feedback. These calculations include the effects of an overpower scram and of both positive and negative temperature coefficients. All other shutdown mechanisms are conservatively neglected. The nominal beginning of core life value of the moderator temperature coefficient is $-1.37 \times 10^{-4} \Delta\rho/^\circ\text{F}$.

FIG. XVI-1-2 PEAK POWER FOR OPEN-HEAD START-UP ACCIDENTS AT THE MAXIMUM ROD WITHDRAWAL RATE, 0.1 PERCENT $\Delta\rho$ PER SECOND.

The maximum ramp reactivity addition rate in the power range could result from a regulator rod failure. This would give a ramp rate of approximately 0.6 percent $\Delta\rho$ /second but only until the total rod worth (0.3 percent $\Delta\rho$) was inserted. Ramps of this rate, initiated at low power, would insert the total worth prior to reaching full power. After complete insertion, the rate of power rise begins to decrease as it approaches the scram level, and the amount of overshoot is small. Therefore, the maximum continuous rod withdrawal rate of 0.1 percent $\Delta\rho$ /second remains the restricting condition.

The reactor response for accidents at 0.1 and 100 percent full power are shown in Figure XVI-1-3 as a function of the rod withdrawal rate. An accident

FIG. XVI-1-3 ROD WITHDRAWAL ACCIDENT IN THE POWER RANGE.

at 0.1 percent full power of 6 times the maximum continuous rate or an accident at full power of 14 times the maximum continuous rate will not cause the power to exceed the beginning of core life burnout level. The effect of changing scram level to 1.2 N_F displaces the curves vertically an equivalent amount.

It is therefore concluded that a rod withdrawal accident with the maximum reactivity addition rate, 0.1 percent/second, is safe for the nominal operating conditions. In addition, a rod withdrawal accident in the power range five times the maximum rate will not produce burnout conditions with any one of the following parameters; 200 percent nominal scram delay time, a one-safety rod scram, or a 75 percent nominal beginning of core life moderator coefficient.

For open-head, low-flow reactor operation, the beginning-of-core-life burnout level is a factor of 2.65 over the operating level. Figure XVI-1-4 shows the peak power for withdrawal accidents occurring at N_L .

2.4 Experiment Feedback

If the experimental loop heat transfer coefficient could be increased to four times the nominal value, the specimen temperature rise would be decreased

FIG. XVI-1-4 OPEN-HEAD ROD WITHDRAWAL ACCIDENT PEAK POWER.

almost 50 percent, but the loop coolant temperature increases only about one degree. The resulting increase in feedback is insignificant. Therefore, specimen geometry or characteristics will only have a small effect on the transient which can easily be compensated by a very small change in fuel region temperature.

3. OTHER SOURCES OF REACTIVITY ADDITION

3.1 Fuel Element Collapse

Fuel plate deflection can occur by pressure gradients developing. Analysis defining the worst possible accident conditions assumes the structural limits of the fuel element have been exceeded followed by the collapse of the fuel element. The reactivity change from this condition has been calculated to be

strongly negative through a range of collapsed areas in both inward and outward directions. Therefore, a reactor power increase cannot occur as a result of this condition.

3.2 Damage to the Beryllium Reflector

Loss of beryllium reflector by damage, followed by replacement with water, results in negative reactivity effects. This result is based on calculations which show that increasing the ratio of water to beryllium results in decreasing reactivity. Therefore, damage and removal of pieces of the reflector cannot cause a positive reactivity excursion.

3.3 Loss of Specimen in the In-Pile Loop

A malfunction of experiment loop equipment could result in the specimen moving in the axial direction and possibly out of the core region. Fuel specimens in the pressurized water loops contribute to the reactivity of the core; therefore, their displacement would result in a decrease in reactivity. Experiments that could cause positive effects by moving out of the core are evaluated by measurements in the ATRC and safeguard review procedures.

3.4 Loss of Reflector Water

Since the water in the reflector coolant channels is a neutron absorbing material, a reduction in the amount of water in the reflector (void formation) is a positive reactivity addition. Voiding of the entire reflector results in +0.71 percent $\Delta\rho$. In the parallel downflow cooling arrangement of the reflector, (multiple holes) flow could be partially restricted by foreign material blocking off the entrance to several cooling channels. The heat generation in the reflector would ultimately fill the blocked channels with steam. The entrance of a small void which has been generated elsewhere in the coolant system will be localized to a small part of the reflector. A large void generated elsewhere will simultaneously enter the fuel region where a compensating negative void coefficient exists. Assuming a large void localized in the beryllium reflector coolant passages surrounding a high powered lobe, the maximum change in reactivity would be approximately 0.16 percent $\Delta\rho$.

3.5 Fuel Handling Accident

Fuel element handling-type accidents have been considered in the core and in the adjacent storage rack. A fuel element fully inserted into its core position would provide a maximum reactivity addition of 1.1 percent $\Delta\rho$. The maximum rate of addition could be 2.2 percent $\Delta\rho$ per second assuming the element is free falling. However, the reactor is designed for refueling with the safety and shim elements inserted to make the core at least 10 percent subcritical; thus, the addition of one fuel element cannot cause criticality.

The k_{eff} of an infinite array of fresh fuel elements surrounded by 1/8 inch of boral (or the equivalent nuclear poison) is 0.70. Thus, the accidental dropping or replacement of a fresh fuel element on the reactor vessel fuel storage rack cannot result in a critical assembly.

3.6 Cold Water Accident

The nominal temperature coefficient for the fuel region is $-1.37 \times 10^{-4} \Delta\rho/^\circ\text{F}$, therefore, a reduction of the coolant temperature results in a positive

reactivity addition. The maximum change and the rate of change in water temperature is limited by the physical arrangement of the coolant system. The primary coolant system consists of one coolant loop with four heat exchangers and four pumps. Only three of the four primary coolant pumps are normally in operation, and the fourth pump is a standby unit. All four heat exchangers will be used at all times. Since the heat exchangers do not have isolation valves, a large, cold stagnant volume of water cannot accumulate in the primary system during operation. Only the pumps have isolation valves, and each pump contains only a small volume of water. Any maldistribution in water temperature, which may exist in the heat exchangers, will be virtually eliminated by mixing in the two common headers through which the coolant must flow before reentering the reactor. Because of the primary system layout, the accumulation of cold water for rapid insertion into the reactor corresponds to less than 0.07 percent $\Delta\rho$. Safety system signals prevent power levels greater than 3 N_L (7.5 MW) unless the primary coolant pumps are operating.

The secondary coolant flow through the heat exchanger is automatically regulated to maintain a constant reactor inlet temperature of 125 °F during power operation. An upper limit on reactivity effects caused by cold secondary water has been calculated on the following basis. The initial conditions of the secondary water are assumed to be a temperature of 32°F and a flow rate of five percent full flow. The accident condition assumes that the flow control valve opens at 3°/second. The flow of 32°F water to the heat exchanger is increased from 5 to 100 percent of full flow. This results in rapid cooling of the primary coolant leaving the heat exchanger. The heat transfer characteristics of the heat exchanger and the heat capacity of the large masses of water limit the rate of change of primary coolant temperature to 1.3°F per second and limit the total temperature reduction to 75°F. This corresponds to a total reactivity addition of 1.03 percent $\Delta\rho$ added at a rate of only 0.018 percent $\Delta\rho$ per second.

3.7 Reactivity from the Experiment

Safeguard review procedures coupled with ATRC reactivity measurements ensure that all experiments to be inserted in the ATR do not compromise reactor safety.

4. EFFECT OF MAXIMUM REACTIVITY ADDITION

Voiding of the high power lobe flux trap (A-1) and experiment loop, instantaneously, provides the maximum reactivity addition accident for the ATR. This accident would require a large rupture of the loop pressure tube followed by rupturing of the insulation can and expulsion of water in the flux trap volume located in the active lattice, all occurring as a step function. Based on experience at the MTR and ETR, rupture of experiment pressure tubes are rare occurrences and do not lead to rupture of the insulating can. The ductility of the materials (stainless steel) prevent fracture-type failures which could cause rapid expulsion of high-pressure water into the flux trap area.

The accident analyzed is, therefore, considered to have a very low occurrence probability.

4.1 Model

The reactor response to step reactivity additions was calculated by using an analog simulation of the reactor kinetics. This model includes the negative and positive reactivity temperature feedback and safety system scram action, but does not include other shutdown mechanisms such as void formation.

Reactivity accident analyses for the ATR were based on a point reactor kinetics model using only average moderator temperature feedback and safety rods to limit reactor excursions. However, the ATR has several different regions, each having a different composition, thermal response, and reactivity feedback effect. The fuel region is the only region in the reactor that has a negative temperature coefficient. All other regions have positive void and temperature coefficients. Because of these basic differences of the ATR from other reactors, the following evaluations were made to determine the specific mathematical model requirements:

(1) A very large fraction of the fission producing neutrons in the fuel annulus come from the reflector. Evaluation of these neutrons as a separate delayed neutron group was not found to be necessary.

(2) The results of the ATR critical experiment showed a significant spacial dependence of the fuel region void coefficient. The outer region or channels of several fuel elements exhibit a positive void coefficient. These have significance in that they are the regions of highest power and, therefore, subject to being the first to produce boiling. However, detailed evaluation showed that the amount of positive reactivity added before the effect became negative could be neglected. The fuel coolant coefficient has a density component which is related to the spacially dependent void coefficient. Since the radial power distribution is similar in shape to the radial void coefficient, the evaluation of spacial dependent fuel plate and coolant expansion effects was initiated to determine the need for multi-channel fuel simulation. The results of this analysis established that single channel simulation would be adequate.

(3) The physical separation of the various fuel regions of the core suggested that the various regions of the core would be relatively independent and would require zone simulation. A series of coupling tests was made in the ATRCE to demonstrate that the different regions of reactor were tightly coupled. The results of these tests show that the point reactor kinetics model will represent the ATR response.

The fact that the various regions of the core are tightly coupled permits the averaging of the reactivity feedback effects of the loops and flux traps. Therefore, two single models were used to simulate the combined effects of nine flux traps and nine experimental loops.

The neck region was not included in the simulation because calculations showed that the time constant for the neck shim housing (a relatively massive structure compared to a fuel element) is approximately four seconds. Although

this region may exert an influence equal to about five percent of the power coefficient, the long time constant (four seconds) makes this feedback effect insignificant for the accidents studied. Accordingly, this region was not included in the mathematical model.

A series of analog studies was made to study the effect of step reactivity insertions. The studies included variation of moderator coefficient, scram delay time, safety rod worth, experiment loop heat, and neutron lifetime. The effect of these parameters on integrated energy was small with the exception of the moderator coefficient. The coefficient used in analyzing the reactivity accident was the nominal value for beginning-of-life conditions. Although the end-of-life value is lower, local power peaking is also lower yielding a greater burnout margin.

4.2 Reactor at Full Power

Measurements in the ATRC have shown that voiding the A-1 flux trap and experiment can produce a maximum reactivity step of +1.05 percent. The integrated energy from the excursion produced by this accident was calculated to be 130 MW-sec as shown in Figure XVI-1-5.

The energy partition between fuel and coolant during a nuclear excursion has been reported as approximately equal based on Spert and BORAX transient data [1, 2]. The fraction of the above integrated excursion energy heating the fuel plates is, therefore, approximately 65 MW-sec.

Since 50 MW-sec will result in melting at the hot spot, a fraction of the core is expected to melt. This calculation assumes no heat to the coolant. The extent of core damage cannot be evaluated until experimental data is available related to recovery from film boiling heat transfer conditions. The affect of this accident on reactor integrity is analyzed under reactor damage.

4.3 Open-Head Operation

The voiding (A-1 flux trap) accident under open-head, low-power operation has been analyzed. Open-head operation, although occurring at a much lower frequency than normal operation, leads to an accident that is more severe than the accident under normal operation because the heat transfer rate from the fuel plates to the coolant has been reduced to approximately 10 percent of normal. The coolant temperature rise and related moderator coefficient reactivity feedback is, therefore, also reduced producing an unacceptable energy release for this accident.

It will, therefore, be necessary to either limit the amount of voiding possible in the A-1 flux trap by increasing the metal water ratio, limit operation of the loop to cold conditions, or reduce the interlobe power split to reduce the maximum reactivity addition to less than one dollar during open-head operation. Figure XVI-1-6 shows the calculated energy release during open-head operation as a function of reactivity insertion. This curve is based

FIG. XVI-1-5 MAXIMUM REACTIVITY ACCIDENT FOR THE ATR AT NORMAL POWER.

FIG. XVI-1-6 MAXIMUM REACTIVITY ACCIDENT FOR THE ATR OPEN-HEAD OPERATION.

on beginning-of-life moderator coefficient, safety rod acceleration of 1.5 g, and scram delay time of 0.025 sec.

The results show that for step reactivity insertions of less than one dollar, the integrated energy of the excursion is less than 4 MW-sec. No core damage will occur since this is below energy additions that caused any core damage during Spert I and BORAX I experiments. Since the ATR core is larger by a factor of two compared to BORAX I, and three compared to Spert I cores, the above direct comparison is conservative.

5. SUMMARY OF REACTIVITY ADDITIONS

The ATR can tolerate the following reactivity accidents without core damage:

- (1) Control element failures
- (2) Start-up accident
- (3) Gauged withdrawal of all rods starting five decades below full power and at full power
- (4) Fuel element collapse
- (5) Damage to reflector and void formation in reflector
- (6) Fuel handling accident
- (7) Cold water accident.

The maximum credible reactivity accident caused by rupturing a high-pressure water loop pressure tube and insulation tube, and instantaneously voiding the flux trap in the high power lobe is expected to cause core damage.

XVI-2. NONNUCLEAR ACCIDENTS

1. LOSS-OF-COOLANT-FLOW ACCIDENT

1.1 General

Examination of the ATR primary system shows the following accidents can cause a loss of coolant flow:

- (1) Loss of commercial power to the 13,800 V main feeder bus
- (2) Failure of one or two 4160 V commercial power buses supplying power to the three primary coolant pumps (The primary pump loads are divided between the two buses.)
- (3) Failure of a primary pump shaft or other mechanical failure
- (4) Closure of the primary flow control valve by operator error
- (5) Closure of the primary system inlet coolant block valves by operator error.

Closure of either the primary flow control valve or block valves requires 53 and 115 seconds, respectively. The valve closure rates are relatively slow compared to flow coastdown rates following pump outage accidents. Therefore, the limiting accidents analyzed were complete power outage causing the loss of all primary pumps and a mechanical failure of one pump.

1.2 Safety Action

The reactor is protected from damage during loss-of-coolant-flow accidents by flow and plant power monitoring. Flow through the reactor is divided into four quadrants having flow measuring instrumentation in each of the four reactor outlet pipes. Each flow instrument has a duplicate backup channel to provide reliability; thus, a total of eight signals and eight scram relays can be actuated during a flow reduction. The response of these instruments, including the recorders that trip switches, is very fast. This response has been specified as being capable of indicating a 10 percent step change in flow in 150 milliseconds. These recorders energize a scram-bus relay and produce a relay scram. The time delay existing after the recorder trips and the safety rods are released is assumed to be 60 milliseconds.

The commercial power supply is monitored on both of the 4160-volt buses that supply the primary coolant pumps. Each bus has an independent monitor to detect a loss of power. The response of the power detection instrumentation has been specified as being able to detect a loss of power within 8 cycles or 133 milliseconds. This provides a faster power reduction than flow monitoring. However, power can be lost to the pumps without loss of the 4160-volt buses. Therefore, the slower response flow monitors have been used in this analysis. Additional protection is also provided by core pressure measuring instrumentation.

Forced coolant flow is maintained at the end of a complete primary coolant pump coastdown by the emergency coolant pumps. The diesel electric powered pump is in continual operation during reactor operation. Since a simultaneous loss of both commercial and diesel power could occur, the battery-backed emergency coolant pump is started on loss of commercial power to ensure the continuation of forced flow through the reactor. The battery-backed pump will be at full speed in seven seconds after loss of commercial power.

1.3 Description of Calculations

The model used in the analog computations is shown in Figure XVI-2-1. This model excluded the pressurizing system and purification system. Mass energy balances for different parts of the primary coolant loop were written and solved simultaneously. The system separation was made as follows: main coolant pump discharge header to pump suction header (reactor vessel, reactor core, reactor inlet, heat exchanger, and main loop throttle valve); and main coolant pump suction header to discharge header (the main pumps and the discharge check valves).

The following assumptions were made: (a) the ratio of the pump flow to pump rotational speed is directly proportional to the ratio of the pump total discharge head to the rotational speed squared; (b) the electrical retardation torque upon loss of power to the pump is negligible; and (c) the friction and windage torque of the pump is small.

1.4 Results

The results of the calculation for the two accidents studied are shown in Figures XVI-2-2 and -2-3. The first case involves the simultaneous loss of electrical power to all three operating primary coolant pumps and the second case concerns the course of events if the shaft of one of the three operating pumps were to break resulting in the assumption that the moment of inertia of that pump is reduced to zero instantaneously.

The moment of inertia reported by the pump supplier was 2084 lb-ft². The value used in the analysis of the accident shown in Figure XVI-2-2 was 1935 lb-ft² leading to a conservative coastdown rate. The system coastdown for loss of all three operating pumps and for the coastdown upon loss of electrical power to one operating pump will be measured during the pre-start-up program and compared with the analog calculations to assure the established setpoint of 90 percent is satisfactory.

The flow rate required to avoid thermal burnout was calculated on the basis that burnout will occur whenever the hot track, exit bulk temperature achieves the local saturation temperature. Local heat flux burnout predicted by the Bernath relationship leads to the same results. Burnout occurs (250 MW reactor power) at 82 percent nominal operating flow. The more limiting case, loss of all three pumps, requires that the reactor power be scrammed in less than 600 milliseconds. Figure XVI-2-2 shows the safety system response time to be adequate. The summation of time lags is: 300 msec coastdown time to 90 percent flow, 50 msec time lag of the flow instrumentation, 50 msec delay time of the two relays, and the scram delay time of 60 msec resulting in an overall delay time of about 460 msec for the power reduction to start. Using a negative 4 percent $\Delta k/k$ insertion (3 rod scram) the reactor power is reduced to 50 percent in 120 msec. An error propagation analysis for heat transfer

FIG. XVI-2-2 EVALUATION OF THE LOSS OF THREE PRIMARY COOLANT PUMPS.

FIG. XVI-2-3 EVALUATION OF THE LOSS OF ONE PRIMARY COOLANT PUMP.

2.1 General

Primary system design is based on conservative engineering practices and applicable codes. The design features and inspection programs specified to ensure the integrity of the reactor vessel and other primary system components have been described in this report. It is concluded that the design, construction, and inspection of the primary system makes it highly improbable that a gross failure will occur.

The core is encircled by a core reflector tank which (a) maintains the water level above the fuel elements, (b) provides a circuit for core cooling by natural circulation, and (c) directs the emergency fire water through the core.

The reactor primary system and auxiliaries can sustain certain sized ruptures without core damage. Analyses of the effect of external leaks at various pressure locations around the reactor pressure vessel and the primary coolant system, and of leaks in the reactor internal outlet flow pipes are presented in this section.

2.2 External Leaks

Failure of components or piping in the pressurized system causes a loss of primary coolant. The consequences of the accident are dependent upon the size and location of the failure and the availability of sufficient makeup water. The reactor primary system has four sources of emergency makeup water: the fire-water system, the normal demineralized water system, the gland seal system, and the flush water system. Only the fire-water system is considered in this analysis. Neglecting the other systems leads to very conservative results since preliminary operational tests indicate that primary coolant pump operation can be maintained, using makeup, for leaks of at least six inch diameter in size.

At the instant the rupture occurs, the reactor and the associated systems in the plant are assumed to be operating normally at full power operation. The reactor inlet pressure is 355 psig and the reactor power level is at 250 MW. Three primary coolant pumps and one emergency coolant pump are in operation.

The reactor inlet pressure is monitored by two independent, electrical pressure transmitters which form a backup pair. The full scale response of these transmitters to a step input is 350 msec. The characteristics of these transmitters are included in the calculation of pressure reduction to determine the time of a power reduction signal. A low-pressure scram setpoint of 335 psig at the reactor inlet was used. The safety rods were assumed to start into the core 60 msec after the transmitter reached 335 psig. This 60 msec is the relay scram delay time.

The criterion for evaluation of the loss of coolant accidents is that steady state burnout power will not be exceeded for the available pressure and flow conditions. During the initial phase of the accident, the flow remains constant but the reactor pressure drops rapidly at a rate dependent upon the leak size and location. This rate establishes the time at which a scram will occur. The pressure required versus the pressure available is computed to determine the maximum size rupture at any location that can be tolerated without reaching burnout. Burnout during this phase of the accident occurs when the hot track, coolant temperature reaches the saturation temperature at the fuel channel exit.

The failure first reduces the primary system pressure, and pump flow remains constant until the pump suction pressure drops below 4.1 psig. Once this pressure has been reached, the flow continues at a reduced rate until pump cavitation occurs. After the primary pumps are unable to maintain flow through the core, the emergency coolant pumps provide flow (4700 gpm with one pump; 5400 gpm with two pumps in parallel) for core cooling. At least one emergency coolant pump will be available to circulate primary coolant until the water level in the reactor vessel drops to the reactor outlet nozzles even in the event of simultaneous (a) loss of commercial power, (b) loss of diesel power, and (c) loss of system pressure.

After the primary pumps become air-bound, the fire-water injection valves open when the reactor vessel level drops to the reactor outlet nozzle level. At this time, approximately 2000 gpm of fire water flows into the core support tank below the core in the reactor vessel.

The results of the evaluation of the accident during the initial blowdown phase (flow remains constant but pressure is reduced) are shown in Table XVI-2-1.

TABLE XVI-2-1

MAXIMUM ALLOWABLE EXTERNAL LEAK SIZE -- PRESSURE LIMITED

Initial Pressure at Leak (psig)	Location in System	Maximum Leak Size (inches)
404	Primary coolant pump discharge	5
355	Reactor inlet lines	5
355	Reactor vessel -- above outlet nozzle	5
255	Reactor vessel bottom head	6
255	Reactor outlet lines	6
230	Primary heat exchanger	6

When the primary coolant pump suction pressure is reduced to 4.1 psia, the gas in the surge tank has expanded from 50 ft³ at 215 psig to 1000 ft³ and is starting to flow out of the surge tank into the primary pump. This causes the reactor flow to decrease from full flow to emergency coolant pump flow. Also, the level in the reactor vessel begins to decrease and the primary system goes to atmospheric pressure (static head neglected). At this time, the accident evaluation becomes dependent upon flow-burnout requirements at atmospheric pressure. A typical evaluation of this portion of the accident is illustrated in Figure XVI-2-4 for a three-inch-diameter leak in the reactor inlet line. The dashed curve is the flow required to prevent burnout, and the solid line is the flow available. The first flow reduction is caused by air-binding of the primary coolant pumps. The second flow reduction is caused by actuation of the fire-water injection valves, which open when the reactor vessel level drops to the centerline of the reactor outlet nozzles (atmospheric pressure in the reactor vessel).

FIG. XVI-2-4 REACTOR FLOW VERSUS TIME AFTER RUPTURE (THREE-INCH-DIAMETER LEAK IN REACTOR INLET LINE, 355 PSIG).

The results of this flow requirement evaluation are tabulated in Table XVI-2-2.

The largest allowable external leak size was three inches at the reactor vessel bottom head, and slightly under three inches at the reactor inlet line. These analyses were performed at the lowest elevations in the system. Accordingly, larger size ruptures can be tolerated at higher elevations. The burnout value for this phase of the accident was based on the factors listed on page 204 and is believed to be conservative for decay heat flux. Experiments currently in progress will verify the calculated burnout value.

2.3 Internal Leak

The internal leak is assumed to occur in the piping inside the reactor vessel due to a mechanical failure in one of the four reactor outlet flow pipes.

The two extreme positions in the reactor outlet flow lines were studied as possible leak locations. These locations were (a) near the flow distribution tank and (b) at the reactor outlet nozzle. Sixteen simultaneous equations for the

TABLE XVI-2-2

MAXIMUM ALLOWABLE EXTERNAL LEAK SIZE -- FLOW LIMITED

Initial Pressure at Leak (psig)	Minimum Elevation of Leak[a] (feet)	Location in System	Maximum Leak Diameter (inches)
404	62.5	Primary coolant pump discharge	3
355	45	Reactor inlet line	slightly under 3
355	86	Reactor vessel above outlet nozzle	3
255	67	Reactor vessel bottom head	3
255	41	Reactor outlet line	3
230	55	Primary heat exchanger	3

[a] Water level at top of reactor vessel is at the reference elevation of 100 feet.

various flows and pressures were solved on a Philco-211 computer for solutions to the two leak locations.

A rupture in the outlet flow pipe opens another parallel path for flow through the core and internals. Different rupture sizes were assumed, and the new steady state flows through the rupture and the other flow paths were calculated. Hence, a rupture bypasses a portion of the system flow around the core, and as the leak diameter size is increased, more flow is bypassed around the core. This reduces the pressure drop across the reactor vessel, increases the flow from the primary coolant pumps, and decreases the flow through all the quadrants in the core.

The analysis showed that rupture sizes up to 8.6 inches near the flow distribution tank will not cause burnout conditions to develop. If the rupture is at the top of the outlet pipe, up to a 9.5-inch rupture will not cause burnout conditions. A high flow scram will occur for a 3.7-inch rupture near the flow distribution tank or a 3.8-inch rupture at the top of the outlet pipe. A setback on high flow would occur at even smaller leak sizes, thus providing ample plant protection.

3. LOSS OF POWER

3.1 Effect of Electrical Power Failures on Operation

The ATR has two independent sources of electrical power: one is from commercial power and one is from diesel-driven generators. The loads carried by each source are shown in Figure XVI-2-5. The diesel power is supplied by two generators, one of which is in continual operation.

FIG. XVI-2-5 ELECTRICAL POWER SOURCES AND LOADS.

Critical plant loads are split between commercial power and diesel power, and some critical loads are supplied from a diesel-commercial bus normally connected to diesel power but automatically transferred to commercial power if diesel power is lost. Therefore, if a commercial power failure occurs, the operating diesel generator continues to supply power to its normally connected loads and to the diesel-commercial bus loads. A loss of either diesel or commercial power will automatically shut down the reactor.

The requirement for maintaining forced coolant flow through the core for several minutes after a shutdown requires power to be maintained to the emergency coolant pumps. The two emergency coolant pumps have independent power supplies and are sized to provide adequate core cooling immediately after a scram and coastdown of the main pumps. One emergency coolant pump is supplied from the diesel-commercial bus and is in continuous operation while the reactor is in operation. The second emergency coolant pump is a standby with dc power supplied directly from batteries. This pump is automatically put into operation when a commercial power failure occurs, when a diesel power failure occurs, or when flow from the diesel-commercial pump is reduced. The battery-backed pump will be running at full capacity seven seconds after one of these conditions is sensed, and is capable of supplying sufficient cooling for a minimum of 30 minutes on batteries alone.

Adequate time is available (minimum of 30 minutes) to start the standby diesel, and it is, therefore, very unlikely that natural convection cooling of the core will be required. In the event natural convection cooling is required, a flow reversal will take place following loss of all forced circulation. The heat transfer relationships are uncertain during this flow reversal period with either nucleate boiling or film boiling a possibility. It is, therefore, possible that some damage in the hot channel could occur.

3.2 Effect of Loss of Power on Shim Controls

The excess reactivity of the ATR (including burnable poison) in the cold clean condition is 9.6 percent Δk . The worth of the safety rods is 6.8 to 7.5 percent Δk . The reactor, therefore, has sufficient reactivity to become critical with the safety rods fully inserted. This situation can only occur during the decay period of an xenon transient (Figure XVI-2-6) following a scram. The following sequence of events is required to result in this situation:

- (1) The reactor has operated at full power and is scrammed. In an attempt to override xenon, the shim rods are withdrawn to or near outer limits.
- (2) Both commercial and diesel power fail.
- (3) Evacuation of the plant is required.
- (4) Plant reentry cannot be made for 55 hours. (See Figure XVI-2-6, xenon decay curve.)

Although the probability of the above sequence of events is low, an additional safety feature has been incorporated in the ATR design. An automatic emergency power source, supplied by batteries, is provided to insert all shim control rods if normal power fails.

FIG. XVI-2-6 XENON TRANSIENT (100 PERCENT TO ZERO POWER).

3.21 The Shim Motor Power Supply System. The ATR shim control element drive motors are normally supplied with three-phase electrical power from the diesel-commercial bus. As a backup to this reliable power source, automatic transfer to the battery-backed, ac power supply for instrumentation occurs in the event of a failure. Both of these sources of power are equipped with phase-reversal detection equipment to prevent the accidental reversing of the shim motors. A schematic diagram of this system is shown in Figure XVI-2-7.

The power for withdrawal and insertion is provided by two separate buses. These buses are mechanically interlocked to prevent simultaneous power for insertion and for withdrawal. Rod insertion is the preferential condition if a choice is required. Since the insertion bus is divided into four sections and the mechanical interlocks are at each motor control, the system could sustain many simultaneous failures before a total insert would fail to occur when required. Because the withdrawal circuit is disconnected from the motors during an insert, this same equipment prevents a continuation of a rod withdrawal accident after a control or safety action has occurred.

Both the withdrawal and insert buses have circuit breakers between the power supply and the motors. Since the power to the motors is distributed in four sections, the breakers provide for the isolation of only one-fourth of the power supply. The effects of the following system failures have been examined.

3.22 Line-to-Ground or Line-to-Line Short. A short to ground on the 208-volt supply line between the section circuit breaker and the transformer fed by the 410-volt, diesel-commercial bus results in the tripping of the circuit breaker which feeds the transformer. All power is lost to the withdrawal bus, and an alarm notifies the operator of loss of normal power to the shim motors. An automatic transfer of the battery-backed, ac instrumentation power to the rod insert buses occurs. This transfer provides assurance that a shim insertion from all points but the regulating position can be accomplished when required.

If the short occurs between the section circuit breaker and a shim motor, the short causes the section circuit breaker to open and one-fourth of the rods are made inoperable. Movement is limited to insertions if the fault occurs between the breaker and the mechanical interlock at the motor in the withdrawal circuitry and vice versa. The one exception is the neck shim drive in the regulating position. This unit is completely disabled by a short in the withdrawal circuitry, but is not affected by a short in the insertion circuitry.

A short in or near the motor results in a total disablement of one-fourth of the shim units (two outer shim motors and six neck rods). This short in one motor causes the circuit breakers to open, and neither an insertion nor a withdrawal of the control elements in this section is possible until the faulty unit is repaired or removed.

3.23 Open Circuit. The location of the open circuit fault determines the number of shim motors affected. If the fault occurs in the withdrawal bus, the withdrawal of rods and insertion from the regulating position is impossible. If the fault occurs in the motor, only withdrawal or insertion of one drive unit is affected.

FIG. XVI-2-7 SHIM AND SAFETY ROD DRIVE POWER SUPPLY.

The loss of the normal insertion bus power (open circuit) results in a transfer to the ac, battery-backed instrumentation power supply. Accordingly, the shim rods can be inserted when required; however, an open circuit between the section circuit breaker and the motors prevents a maximum of one-fourth of the shim elements from inserting (with the exception of the element in the regulating position). This situation could arise if a section circuit breaker is left open. Interlocks are provided that prevent rod withdrawal if any circuit breaker is open or voltage to the bus is low. In addition, a scram will occur if any of the outer cylinder control rod circuit breakers are open or low voltage across the bus occurs.

3.24 Phase Reversal. Phase reversal equipment is utilized on both the diesel-commercial supply and the battery-backed supply to the shim rod drive motors. Should a phase reversal occur on the normal insert circuit, this circuit would be automatically transferred to the battery-backed supply, and an alarm would notify the operator of this condition. The withdrawal bus would not be interrupted because a signal to withdraw rods would insert them if a phase reversal occurred.

If the phase reversal occurs on the ac, battery-backed supply, this supply is removed from the circuit. An alarm notifies the operator of this condition.

3.25 Conclusions. The safety of the ATR plant requires that a reliable shim insertion system be used. To provide this system, the insertion mode must be independent from the withdrawal mode except at the motor. A third power supply is provided for the ATR shim rod insertion mode. This addition, plus dividing the total shim rod worth into four groups to minimize the effects of possible line faulting, greatly enhances the safety of the plant. All of the safety rods plus 50 percent of the shim rods would remove 10.4 percent Δk from the core. This would be adequate to maintain the reactor subcritical with all xenon decayed from the core.

Adequate time is available to start the standby diesel and to restore normal power to the shim drives. The diesel units can be started from the process control room, which is outside the ATR gastight building, and they could be started in the event that the remainder of the plant required evacuation.

It is, therefore, concluded that an improbable number of failures would be required to prevent a shim insertion when needed.

4. FUEL ELEMENT FAILURES

A fuel element cladding failure releases volatile and nonvolatile fission products to the environment. The consequences of such an accident depend upon the condition of the fuel and the environment in which the failure occurs. Should the failure occur in the closed primary system, contamination would be limited to the primary system; however, should the failure occur during fuel transfer or in the canal area, more widespread activity would result.

4.1 Fuel Element Failures in the Primary System

MTR and ETR operations have shown that fuel element cladding failures are more of a nuisance than a hazard; such failures in the enclosed primary

system have not resulted in an undue hazard to personnel or to the environment. The most probable cause of in-pile failures is flow restrictions resulting either from distortion of the fuel plates or from foreign material. Many distortions can be prevented by proper inspection and handling procedures. Cleaning and operating procedures to ensure removal of foreign material that could restrict the coolant flow in the fuel elements can also prevent incidents.

The ATR fuel elements have been hydraulically tested in an extensive development program to ensure design adequacy. Each production element for the initial core loading will be flow tested and inspected to ensure structural integrity before loading into the reactor. (The frequency or number of elements per core to be hydraulically tested will be determined after obtaining experience with the behavior of several cores.)

An evaluation was made to determine radiation levels outside the primary piping shielding for an accident involving the release of fission products to the primary coolant. The accident released to the coolant the gross fission product activity in one fuel element after 17 days of operation. It was assumed that fission products were dispersed uniformly throughout the coolant volume. The dose rate from a 36-inch pipe through a 5.5-foot concrete shield two minutes after the accident was calculated to be 5 mrem/hr. During normal power operation, the maximum dose rate from N-16 activity in a 36-inch primary pipe through equivalent shielding was computed to 7.5 mrem/hr. Of all areas normally accessible during operation, the area outside the pump cubicles has the highest radiation level due to such an accident. The pumps are shielded by four feet of ordinary concrete, and the dose rate at the outer surface of these walls two minutes after the accident is estimated to be 150 mrem/hr. The resulting dose rate from a fuel element failure is less than that from the normal N-16 activity of the primary loop with the exception of the pump cubicle walls. Therefore, the normal shielding is more than adequate to protect against radiation that might be released from fission products, and no direct radiation hazard to personnel exists.

Detection of a fuel element failure in the primary system is provided by a special fission break monitor installed on the outlet piping of the reactor. Upon receipt of a fission break signal, a relay scram occurs.

To protect the reactor against excessive flow which could collapse a fuel element, a series of redundant reactor pressure drop measurements is used to scram the reactor in addition to shutting off one operating pump.

4.2 Flow Blockage Accident

Fuel element flow blockage accidents can occur by relatively large size extraneous material entering the reactor because of equipment failure, corrosion during operation, and error during shutdowns. The design requirements, equipment quality, and inspection procedures for the ATR minimize the possibility of equipment failure and excessive corrosion. Administrative procedures developed at the MTR and ETR for shutdown tank work will be applied to the ATR. These procedures, to minimize extraneous material inadvertently entering the reactor, are based on years of successful shutdown work and also several errors that have occurred.

The ATR core arrangement and fuel element design tend to prevent the blockage of large areas of the core. In addition, fuel element side plate vents will provide a secondary flow path for the coolant.

Detection of flow blockage of small areas of the core is by the fission break monitors.

It is concluded that a flow blockage accident can occur in the ATR. Design and administrative control procedures minimize the possibility of this accident and restrict the accident to small areas of the core.

4.3 Cladding Failures During Fuel Element Handling Operations

In the ATR the fuel elements are completely covered with water during all phases of transfer to provide for natural circulation cooling and shielding. The fuel elements are grappled and locked to a fuel element handling tool when they are raised out of the core and transferred to the drop tube. During all transfer operations, the fuel element is maintained in a vertical position to ensure adequate natural circulation and cooling. The drop tube contains an insert which permits only one element to be transported at a time. All storage racks are designed to prevent a critical mass even if an additional fuel element is accidentally dropped on the top of the storage rack. Accordingly, the ATR fuel handling procedures are designed to prevent cladding failures from overheating or criticality accidents.

5. REACTOR MISSILE PRODUCTION

The missile potential of structures, such as the top shield ring and transfer shield plate which are not solidly anchored to the reactor or the building, has been investigated.

5.1 Reactor Top Shield as a Missile

A pressure wave generated by a large energy release in the core will strike the upper bolted closure and will be reflected from it even though the reactor vessel remains intact. When this occurs, an impulse is imparted to this structure. Because the shield ring and plate are not anchored, any velocities imparted to these components would cause them to rise. The total impulse imposed by a shock wave can be determined by using data from the Ballistics Research Laboratories (BRL) for measured impulses of Pentolite explosions underwater [3]. This impulse can be imposed on the shield ring only as long as the elastic stretch of the studs and vessel will permit. This limit is conservatively neglected.

The BRL data relates specific impulse with the distance from the explosive charge for underwater explosions of Pentolite.

$$\frac{I}{W^{1/3}} \text{ is a function of } \frac{R}{W^{1/3}}$$

where

I = specific impulse = impulse per unit area perpendicular to shock wave (lb-sec/in²)

R = distance from charge (feet)

W = weight of Pentolite explosive (pounds).

To determine the height the shield ring would rise, the relationship between impulse and imparted velocity must be established.

$$I_t = \int F(t) dt$$

$$V = \int a(t) dt = \int \frac{F}{M}(t) dt = \frac{I_t}{M}$$

$$I_t = MV$$

where

$I_t = IA$ = total impulse on reactor head

F = instantaneous force on head = Ma

V = imparted velocity to head and shield ring

a = instantaneous acceleration

M = mass of head and shield ring = W/g

A = projected area of vessel head.

The shield ring, shield plate, and head have a combined weight of 164,190 pounds.

$$I = \frac{I_t}{A} = \frac{MV}{A} = 0.3295 V$$

The energy imparted to the shield ring and plate is given by

$$\begin{aligned} KE &= 1/2 M V^2 \\ &= \frac{W V^2}{2g} \end{aligned}$$

where W = weight of shield ring and plate (113,940 pounds).

This kinetic energy is transformed into potential energy as the shield ring and plate reach the top of their trajectory.

$$P. E. = Wh$$

Equating these two forms of energy,

$$Wh = \frac{W V^2}{2g}$$

$$V = \sqrt{2 gh}$$

$$I = 0.3410 \sqrt{2 gh} = 2.740 \sqrt{h}$$

Table XVI-2-3 shows the vertical displacement of the shield ring as related to energy release.

TABLE XVI-2-3
TOP SHIELD DISPLACEMENT

<u>Energy Release (MW-seconds)</u>	<u>Specific Impulse (lb-sec/in²)</u>	<u>Equivalent Pounds of Pentolite</u>	<u>Vertical Displacement (feet)</u>
50	0.994	26.3	0.13
75	1.29	39.5	0.22
100	1.53	52.6	0.31
150	2.02	78.9	0.54
200	2.37	105.2	0.75

5.2 Transfer Shield Plate, Shield Plug, or Loop Closure Plug as a Missile

The top seal plug of a pressurized water loop is a four-inch-diameter closure plug, designed so that it is one of the strongest points in the loop system. It is covered by the loop transfer shield plate during normal operation. An analysis was made of the missile potential of the loop transfer shield plate from the impact of a loop closure plug and from pressure buildup under the plate.

The velocity of the loop closure plug was computed after an assumed failure with the loop at pressure. This velocity was 220 feet per second at the time of impact with the transfer shield plate and the shield plugs. The weight of the loop closure plug was assumed to be 44 pounds (the weight of a steel cylinder 1 foot long and 4 inches in diameter). A momentum balance to determine the velocity of the transfer shield plate was made.

$$V_{TS} = \frac{V_{SP} M_{SP}}{M_{TS}}$$

$$V_{TS} = \frac{220 (44)}{31,940}$$

$$V_{TS} = 0.30 \text{ ft/sec}$$

where

V_{TS} = velocity of Transfer Shield Plate, ft/sec

V_{SP} = velocity of Loop Seal Plug, ft/sec

M_{TS} = mass of Transfer Shield Plate, pounds

M_{SP} = mass of Loop Closure Plug, pounds.

The maximum height in the transfer shield plate trajectory was computed by a balance between the kinetic and potential energy. The maximum height in the trajectory is 0.003 inch.

The pressure buildup of the steam vented from the loop under the shield plate cannot raise the plate from its support because the chamber under the transfer shield plate is vented through the special ventilation ports. Since the plate is locked to its support, the steam pressure required to lift the shielding is ≈ 7.5 psig. The maximum calculated pressure buildup for the rupture of the Conax Seal is 1 psig. Accordingly, shielding lift will not occur.

The shield plug missile potential was also examined. Taking the loop closure plug as the projectile, the resulting velocity of the 160-pound shield plug is 16.9 feet per second. The maximum height of the trajectory is 4.4 feet. To eliminate the missile potential of the shield plug, the shield plugs will be held in place by a locking plate.

5.3 Production of Missiles from Reactor Internals

The relationship of missile production and core damage to energy release in the ATR is assumed to be comparable to that in BORAX I and Spert I. The volume of the ATR core is approximately two times that of BORAX I and three times that of Spert I. Therefore, direct comparison of damage on a MW-sec basis should be very conservative. The degree of damage associated with various energy releases in BORAX I is reported in AECD 3840; Spert I, in IDO-16885.

An energy release of 4 MW-sec in BORAX I produced a chugging action and splashed some water out of the vessel. Energy releases of 9 MW-sec expelled large volumes of water from the reactor vessel. In the energy range of 9 to 26 MW-sec, water was expelled, and fuel plate distortion occurred. Although the elements were deformed, there was no evidence of gross mechanical destruction of the fuel plates or missile generation. During the final BORAX I excursion, a release of 135 MW-sec destroyed the core and ejected core and

reactor parts from the vessel. The BORAX I experience indicates that no missiles are produced during excursions of less than 26 MW-sec. On this basis, the ATR fuel elements will remain intact, and since they are the most fragile component in the core and are in the region of maximum pressure and accelerating forces, it is reasonable to expect that missile formation will not occur. In the Spert I tests, 7.7 MW-sec caused fuel plate distortion, and 17.5 MW-sec caused melting of some of the fuel plates. The final test which destroyed the core, released 30.7 MW-sec.

This Spert and BORAX data may be used to predict the onset of fuel damage in the ATR. To be conservative, this data will be used for comparison with the ATR without any volume correction.

5.31 Effect of Core Missiles on the Closed Reactor Vessel. A calculation was made to determine the range of missiles in the reactor vessel water by using the method presented in ANL 5948 [4]. Assuming that a beryllium missile leaves the reflector area at sonic velocity, the velocity after one foot of travel in water is only five feet per second. The energy of the missile is dissipated before reaching the reactor vessel wall since the reflector is approximately three feet from the reactor vessel. Therefore, missiles generated in the core or the reflector do not compromise the reactor vessel integrity.

5.32 Effect of Core Missiles on the Reactor Room. During normal operation, the reactor vessel is closed and the top shield assembly is in place. However, it may be desirable to operate the reactor occasionally at low power with the handling ports open (as for flux mapping).

On the basis of BORAX I, no core missiles are expected to be generated if the nuclear energy release is less than 26 MW-sec. The maximum energy release in the ATR during open-head operation is less than 4 MW-sec.

6. FUEL TRANSFER ACCIDENT

During the handling of irradiated fuel, the decay heat can cause melting if fuel is removed from the coolant to the air. The plant is designed so that reactor fuel is transferred under water at all times. Experiment samples are transferred in water filled casks. The fission product inventory in any of the experiment specimens does not exceed one percent of that in the reactor core.

The fuel element transfer accident analyzed in the hazards section assumes that the hottest fuel element is accidentally removed from the water and melts. Since the two outside plates can transfer heat to the room, only the center fuel plates are assumed to melt and release fission products. This is an equivalent release of three percent of the total core inventory.

7. PRESSURIZING SYSTEM FAILURES

The ATR uses a dual pressurizing system consisting of the surge tank and a pump and bleed system previously described.

7.1 Air Inventory Accidents in Surge Tank

When the pressure is reduced to atmospheric pressure in the surge tank, the 50 ft³ of air at 215 psig expands to fill the surge tank. This relationship establishes the maximum amount of air that should be in the surge tank at any one time because it is very undesirable to have the primary system depressurize to permit this air to expand into the primary system. To prevent an excess of air in the surge tank, all air inventory adjustments, either admission or discharge, are made under manual control. Double control valves with a bleed to the atmosphere between valves are used to supply air to the surge tank. In addition, a low level switch shuts the air admission valve automatically to prevent excessive air inventory, which could overflow from the surge tank into the primary system. Accordingly, both an operator error and an equipment failure would be required before the surge tank could be charged with an excessive air inventory.

The reactor is protected against an inadequate air inventory by a high surge tank level scram and a high pressure scram.

7.2 Failure of Surge Tank Nozzles and Penetrations

The surge tank has only a few penetrations or nozzles for vent and drain lines and instrumentation connections. The maximum sized nozzle, excluding the six-inch-diameter surge line, is the two-inch-diameter vent line located at the very top of the surge tank. Since this connection is located in the gas volume region, the outflow of air from a failure in this region is limited by the critical pressure ratio. (The evaluation of potential failures located in the liquid region is discussed under loss of coolant accident.)

An air loss due to a failure in the surge tank results in a decrease in reactor inlet pressure to 140 psig. Since the failure is in the gas region, no primary system coolant is lost, and the primary coolant pumps continue to deliver full flow.

The reactor can maintain as much as 55 percent of full power when the reactor inlet pressure is 140 psig. However, a low pressure scram occurs and reduces the reactor power to decay heat levels.

This accident has been evaluated for several rupture sizes at the top of the surge tank to determine the maximum sized rupture that could occur without producing burnout conditions. Figure XVI-2-8 shows the minimum pressure margin as a function of rupture size for release of air from the surge tank. The minimum pressure margin is defined as the actual pressure at the reactor inlet minus that pressure required to prevent burnout. The results of this analysis show that a 6.2-inch rupture could be tolerated without producing burnout. The maximum sized penetration in this region of the surge tank is only two inches in diameter.

7.3 Failure of Pressurizing Pump System

The loss of commercial power causes a solenoid valve to close the pressure control valve, PCV-1-1. This action stops all flow to the degassing tank in four seconds and maintains the system inventory and pressure. The gland seal pumps, powered from the diesel-commercial bus, continue to operate even though a commercial power failure could make the primary pressurizing pumps

FIG. XVI-2-8 EVALUATION OF SURGE TANK GAS RELEASE.

inoperable. These gland seal pumps deliver approximately 125 gpm to the primary system to prevent contamination of seals and other penetrations in the primary system. These pumps restore the reactor pressure lost from the decrease in primary coolant pump head. The cut-off head of these gland seal pumps is approximately 415 psig, which exceeds the setpoint (379 psig) for the primary system relief valves. To raise the entire system from the surge tank pressure (215 psig) to the relief pressure requires an addition of only 157 gallons of water or slightly more than a one-minute operation of the gland seal pump. A high pressure alarm indicates this condition, and the operator may stop or reduce the gland seal flow to prevent overpressurization. The relief valves protect the system if the operator fails to take action.

These conditions require a manual reset of the pressure control system prior to restarting the primary coolant pumps. Failure to do so could cause overpressurization of the primary system and lifting of relief valves.

A failure of diesel power could produce a loss of the pressurizing pump since the pump may be supplied from either power source at the option of the operator. In the event of a power failure to the pump, the pressure control valve which regulates the flow to the degassing tank closes as the pump flow decreases and the system pressure drops. Conservatively neglecting the inflow from the gland seal pumps, the primary system loses 300 gpm or less. A failure of the pump or pump motor from other causes is expected to produce the same effects as previously mentioned.

The pressure control valve, PCV-1-1, is a pneumatic valve which regulates the blowdown flow to the degassing tank to maintain constant reactor inlet pressure. This valve closes upon failure of commercial power or instrument air or upon loss of a control signal. The stroking time for this valve is four seconds, and the net increase of coolant inventory during this stroking time is less than 20 gallons. This action is not sufficient to produce a high pressure (375 psig) scram. However, unless the valve, FCV-1-8, closed rapidly, the reactor coolant inventory rapidly increases to produce a high pressure scram.

Although these failures cause PCV-1-1 to close, a failure which would force the valve to full open has been examined. A maximum flow of 1000 gpm to the degassing tank would result in producing a low pressure (335 psig) scram in less than six seconds (with less than a 100-gallon loss of coolant). The rate of pressure reduction is approximately 10 psi per second. This reduction is well within the capability of the reactor and safety system.

7.4 Conclusions

Examination of various failures in the primary pressure control system shows that neither a hazardous condition nor fuel damage is produced. This evaluation includes the effects of power failures, valve failures, and surge tank ruptures. A surge tank rupture of 6.2 inches in diameter in the gas area can be tolerated without fuel damage. This size is a factor of three over the maximum size nozzle located in the surge tank, excluding the six-inch surge line.

8. EXPERIMENT FAILURES

8.1 General

The ATR facilities provide for nine experimental loops in and around the reactor core. This section analyzes facility failures that could affect the reactor. Specific experiments are reviewed and approved by procedures outlined in Section XV.

8.2 Safety Evaluation of the Loop Closure Plug

The pressure tubes penetrate and are supported at the reactor top closure plate. All loop components are inside the reactor vessel or are located in the basement area of the reactor building with the exception of the portion above the top closure plate. This portion extends into the main reactor operating room

floor. A closure plug closes the top opening of the pressure tube. A failure of a pressurized loop in the region of the top-head area would release a large quantity of pressurized water, or a mixture of steam and water, to the reactor operating room floor. An instantaneous release of a large quantity of this water would cause pressurization of the gastight building.

An analysis of the effects of various rupture sizes has shown that a 0.5-inch-diameter rupture ($\approx$ Conax Seal Diameter) would not pressurize the reactor building. The loop top closure plug has a diameter of about four inches, and a failure would result in an extremely rapid release of the loop contents. Although it cannot be shown that this failure is incredible, the factor of safety against a failure due to internal pressure loadings has been calculated to be not less than seven on any component.

In addition, the required torque to unlock the closure plug locking ring has been calculated to be 550 foot-pounds with an internal loop pressure of 2200 psig.

The largest practical rupture which could be assumed is the diameter of the Conax seal for the specimen thermocouple, approximately 0.5 inch. This will not cause pressurization of the reactor building. This rupture will produce less than 1 psi overpressure in the top-head shielding while approximately 7.5 psi is required for lifting. Therefore, no missiles will be generated from a failure of the loop top closure.

8.3 Evaluation of Loop In-Pile Pressure Tube

In the ATR, the maximum potential reactivity accident has been specified as that which will be caused by the voiding of a flux trap, a pressure tube, or a combination of both the flux trap and the pressure tube. Since this voiding condition is the most severe nuclear accident expected for the ATR, the design of the in-pile portion of the pressure tube has received additional consideration to ensure the integrity of both the pressure tube and the insulation jacket. The insulation jacket surrounds the pressure tube and provides the second barrier between the high temperature loop water and the reactor.

The high temperature water in this assumed accident flashes to steam at 380°F to create the void. This rupture is considered highly improbable since calculations and test reactor operating experience have shown that the failure of the insulation jacket does not follow a pressure tube failure^[5]. The steam released to the annulus between the insulation jacket and the pressure tube is vented to the atmosphere, and full loop pressure is not applied to the insulation jacket. The insulation tube, however, has the ability to contain the maximum loop pressure. Pressure tubes are of ductile material; therefore, brittle fracture producing missiles which will penetrate the insulation jacket are not anticipated. Although the loop pressure exerted in the insulation jacket exceeds the yield strength, the jacket enlargement would not be enough to close the 0.5-inch clearance between the jacket and the safety rod.

8.4 Gas Loop Failure

Although the gas loop design has not been completed, the in-pile section design has progressed to where an evaluation of hazards to the ATR can be made. The loop pressure is maintained below reactor pressure at all in-pile locations. A failure of the loop pressure tube will, therefore, allow reactor

coolant to flow into the pressure tube. This will cause a negative reactivity effect. The gas loop is not surrounded by a safety rod and, therefore, cannot interfere with the reactor safety system.

9. MISCELLANEOUS ACCIDENTS

9.1 Relief Valve Stuck Open

This accident would allow a leak rate of 600 gpm with the reactor at normal pressure. This leak is not as severe as analyzed in the loss of coolant accidents and can be handled by makeup water.

9.2 Earthquake

The ATR vessel structures, piping, and equipment have been designed to conform to Zone 2 (NRTS area) requirements. The seismic detection system will scram the reactor in the event a serious disturbance occurs.

XVI-3. REACTOR DAMAGE EVALUATION

1. GENERAL

Fuel melting accidents are of prime interest not only because of fission product release, but also because the fuel may react chemically. There are two possible chemical reactions that may occur with the ATR fuel plates. These are the "thermite" reaction of the U_3O_8 and aluminum and the metal-water reaction of aluminum and reactor coolant. Both of these reactions are exothermic and, therefore, must be considered in relation to fuel melting accidents.

Three types of fuel melting accidents have been postulated for the ATR.

- (1) A large nuclear excursion can rapidly heat the fuel to very high temperature. At the beginning of core life, a nuclear energy release of 50 MW-sec would produce molten fuel at the hot spot if adiabatic heating of the fuel is assumed after steam blanketing occurs.
- (2) A large rupture of the primary system can produce steam blanketing of the fuel elements possibly leading to fuel melting.
- (3) Fuel melting may be the result of a restricted coolant channel from either physical damage to the element or blockage from extraneous matter in the system.

The maximum energy release from a complete thermite reaction of the fuel matrix is 92 MW-sec. (This amount is sufficient to adiabatically heat some of the fuel areas from the melting temperature to the threshold temperature for an aluminum-water reaction, 1170°C [6].)

The "thermite" reaction is characterized by a rapid initial reaction followed by a very slow reaction rate of less than 10 percent per minute. The initial reaction is referred to as the equilibration reaction and is the more important reaction, because reaction rates are faster; the conversion ranges from a negligible amount at 1400°F to 66 percent at 2000°F. The maximum reaction rate during the equilibration period appears to be about 20 percent U_3O_8 per second based on unpublished ANL data. Explosive reaction rates for ATR fuel composition have not occurred.

A complete metal-water reaction of all the aluminum in the fuel plates (neglecting areas not in contact with fuel matrix) could produce a maximum energy release of 3740 MW-sec. In view of the fact that fuel melting accidents are possible, and the energy releases of the potential chemical reactions of the fuel are large, a more detailed analysis of the reaction kinetics must be used to establish the relative safety of the ATR should fuel melting occur.

Each of the three fuel melting accidents has been evaluated using experimental data from the thermite reaction program conducted at Georgia Institute of Technology on the ATR fuel composition and from the experimental data on metal-water reactions conducted by Argonne National Laboratory [7]. These data show that neither of these chemical reactions produce high order explosions,

are not always self-propagating, and are rate controlled. The effect of these reactions on the compounding of fuel melting accidents may be assessed using these data.

2. FUEL MELTING DURING A NUCLEAR EXCURSION

The duration of a severe nuclear excursion in the ATR would be very short, on the order of a 100 milliseconds, and a scram rapidly reduces the power to decay heat levels. A 50 MW-sec energy release is required to produce molten fuel at the hot spot. This amount assumes adiabatic heating of the steam blanketed fuel region. Beyond this amount of energy release, a thermite reaction can be expected to occur.

A 50 MW-sec nuclear burst would adiabatically heat the hot spot to the molten temperature, 1140°F. The thermite reaction accompanying this burst would be nil. A 100 MW-sec nuclear burst would heat the fuel at the hot spot to 2400°F, a temperature at which aluminum-water reactions could be expected. Assuming a maximum thermite reaction rate of 20 percent per second, a total available core thermite energy release of 92 MW-sec, and a nuclear burst of 100 msec duration, the net result is 1.84 MW-sec of "Thermite" energy associated with the nuclear burst of 100 MW-sec. Since this is only two percent of total heat release, it is considered insignificant when evaluating the probability of the thermite reaction enhancing a metal-water reaction in the ATR.

The relative safety of the ATR with respect to metal-water reactions during a nuclear excursion may be established by a comparison of the maximum postulated accident with the amount of predicted metal-water reaction that could occur even if the accident were much larger. Recent experiments have shown that nuclear excursions can initiate a metal-water reaction, but the reactions that have occurred are neither self-propagating nor high order explosions. This has been demonstrated by the Spert I Destructive Test, the SL-1 incident, the experimental determinations of the kinetics of aluminum-water (steam) reactions at Argonne National Laboratory, and nuclear excursion induced metal-water reaction experiments with the TREAT reactor.

A metal-water reaction occurred in the recent Spert I Destructive Test. The evaluation [8] of this test states that approximately 35 percent of the core melted with all plates experiencing some degree of melting. With this large amount of melting, only 0.4 percent of the amount of aluminum in the core underwent reaction. In addition, the metal-water reaction did not appear to be responsible for the destructive pressure pulse that occurred in the reactor. Fuel temperatures in excess of 1200°C were calculated for the reactor and yet the reaction was quenched, not propagated.

Experience from the Spert I Destructive Tests has shown that a total metal-water reaction of the order of 0.5 percent may be tolerated without significantly compounding the effects of the nuclear excursion. Thus, when the hot spot fuel volume experiencing a high power peak is only a few percent, a rather large reaction could be tolerated in these regions as long as the integral over the core volume remained small.

The same type of phenomenon experienced in the Spert I test was experienced in the SL-1 incident [9]. Over 50 percent of the plate area of the core reached the molten state. A maximum specific energy of 500 calories/gram was estimated for the hottest region of the core and yet only 1.3 kg of aluminum reacted or approximately 1.5 percent of core aluminum, as defined by the heat transfer area.

In addition to the Spert I and SL-1 nuclear excursion induced metal-water reactions, a series of experiments has been conducted in the TREAT reactor to determine the amount of metal-water reaction as a function of the specific energy input to the specimen. In these experiments, a sample is heated very rapidly by a nuclear burst in the TREAT reactor. The specimens are submerged in water at 25°C in an autoclave and the total amount of reaction as function of energy input was determined. Tests have been performed on Spert 1-D plates, SL-1 unclad plate, and Al-U₃O₈ unclad cermet rods.

Spert 1-D plate specimens:

0.060 inch x 0.5 inch x 1.4 inches
0.020 inch clad and 0.020 inch fuel of 77 weight percent Al

SL-1, Unclad plate specimens

0.20 inch x 0.5 inch x 0.5 inch
81 weight percent aluminum

U₃O₈ -- Al Cermet rod (unclad)

0.375 inch diameter x 0.5 inch long
90 weight percent Al and 10 weight percent U₃O₈

The results of these tests are shown in Figure XVI-3-1. The percent metal-water reaction is plotted as a function of the specific energy input to the specimen. It should be noted that all three specimens produce reactions of approximately the same magnitude although there exists a large difference in surface to volume ratio. It is, therefore, concluded that this data would be representative of the ATR fuel plate geometry (0.050 inch thick) and fuel composition (41 weight percent U₃O₈-59 weight percent Al). Additional tests were performed on the SL-1 and Spert plates with the fuel initially submerged in saturated water at 285°C. Total reaction under these conditions is identical for specific energy inputs of 400 calories/gram or less while those at > 500 calories/gram show a much larger reaction. This is assumed to be caused by the effect of boiling heat transfer on the temperature rise of the specimen.

Two clusters of three Spert 1-D plates each were also tested and the results of these tests show that approximately the same amount of reaction occurs with the three plate clusters as do the single plate specimens [10].

The TREAT excursion experiments simulate the rapid heating of the fuel plate that would be experienced in the event of a nuclear accident (periods in the range of 10 to 40 milliseconds). Calculations assuming an adiabatic temperature rise have been made on the Spert I Destructive Test and the SL-1 incident. The calculations agree very closely with the measured (Spert I) or

FIG. XVI-3-1. COMPOSITE PLOT OF TREAT EXPERIMENTAL DATA, ALUMINUM-WATER REACTION.

estimated (SL-1) peak temperatures in the region of maximum power generation. It is, therefore, assumed that the prediction of ATR fuel temperatures, and, hence, the metal-water reaction characteristics, could be based upon adiabatic temperature rise.

The following assumptions were used:

- (1) An initial power density distribution based on a maximum radial peaking value of 2.3 and a cosine axial flux distribution. Later measurements of radial peaking showed a value of 2.66 covering a smaller area of fuel.
- (2) A linear temperature interpolation exists between the TREAT energy input corresponding to fuel melting (220 cal/g at 640°C) and the marked increase in the extent of reaction corresponding to the input of 750 cal/g (1750°C).
- (3) The reactor period will be sufficiently short that adiabatic heating will result. The temperature-energy input correlation was that reported in Reference 5 by ANL personnel.

The most pessimistic correlation of the data of the amount of metal-water reaction for a given energy input (or temperature) was used as shown on Figure XVI-3-1. All data points fall on or to the right of this curve.

The analysis was performed assuming the entire reactor to be at 400°F prior to the nuclear excursion, a very pessimistic assumption.

The results of this analysis are shown in Figure XVI-3-2. The energy release for the maximum reactivity accident in the ATR is 130 MW-sec. This corresponds to a metal-water reaction of less than one percent.

3. FUEL MELTING RESULTING FROM A LARGE PRIMARY SYSTEM RUPTURE

The time scale for fuel melting accidents that are the result of flow or pressure disturbance are much longer than those associated with nuclear excursions (seconds instead of milliseconds). The fuel plate would steam blanket, and melting would occur producing a mass of molten metal held in place by the remaining structure of the element or flowing out of the core into the lower flow distribution chamber to be cooled. In any event, there will be molten metal in contact with steam, and some mode of cooling exists for both the molten material and those portions of the fuel plate which are not in a region of high enough heat generation to produce complete melting.

The effect of a thermite reaction on a rupture accident would depend upon the maximum temperature that the fuel plate would reach before cooling is restored.

The cermet fuel plate exhibits a molten strength which could affect coolant channel geometry during a melting accident. This phenomenon could enhance the possibility of higher fuel temperatures as this would tend to resist movement of molten fuel.

FIG. XVI-3-2 ATR ALUMINUM-WATER REACTION.

If fuel temperatures in excess of 1400°F were to occur in this fuel melting accident because of the molten strength, then a thermite reaction would contribute to the heating of the fuel plate. The thermite reaction could release the thermite energy at a rate of approximately 20 percent per second. The peak temperature attained would depend upon the ability of the fuel to transfer the additional chemical energy as well as the decay heat. The nuclear energy (decay heat) at the hot spot (10 percent of full power) release rate is a factor of four greater than the thermite chemical energy release rate. On this basis, the peak temperature is more dependent on nuclear energy than on thermite energy. Accordingly, high metal temperatures, if they occur, should primarily be associated with the nuclear heat. The thermite energy is only a secondary effect which promotes the probability of achieving the temperature required for metal-water reactions. The probability of reaching fuel temperatures in excess of 2000°F during a large rupture accident cannot be determined.

The physical conditions of the fuel element during a melting accident as described on the previous page may be considered analogous to the experimental work conducted at ANL.

Argonne National Laboratory has done studies on the kinetics of the aluminum-water (steam) reactions which can be applied to slow melting accidents (those melting accidents not associated with a nuclear excursion) in the ATR. The reaction rates have been determined for a 8 mm ball of aluminum that has been levitated in air and in steam at temperatures from 1200 to 1750°C [11]. The results are shown in Figure XVI-3-3. The isothermal reaction is rather slow, taking minutes rather than milliseconds to obtain any appreciable reaction.

The study [9] reports "that at 1600°C and above, the pellets ignited and burned to 90 percent completion in steam. At 1750°C, ignition started nearly instantaneously, whereas at 1700°C, ignition was delayed for 5 minutes. At 1650 and 1600°C the delays were 9 and 18 minutes, respectively. The nature of the ignition and burning processes indicated that ignition occurred when aluminum vapor was able to penetrate the oxide film. Burning then took place between the gaseous aluminum and water vapor". Subsequent unpublished ANL data showed that a maximum rate for this reaction is approximately 1 g/cm²/min. A 5 mm sphere would require 10 seconds to completely react.

It is concluded from this data that below 1600°C an aluminum steam reaction would require minutes, and that above 1600°C where ignition occurs, many seconds are required to obtain a nearly complete reaction.

Should the required temperature for an Al-water reaction be reached in a loss of coolant accident, the subsequent reaction that does occur will be a controlled reaction, not an explosion. The metal water reaction data by the levitation method have been applied to the ATR for temperatures in range of 1200 to 1500°C. At a peak temperature of 1200°C, the ratio of nuclear (decay heat at 10 percent full power) to chemical energy release is approximately 700 to 1. At 1500°C this ratio decreases to approximately 400 to 1. It is not until the ignition temperature (> 1600°C) is reached that a significant chemical energy release would result.

If ignition occurred, the chemical energy release rate would approach that of full power reactor operation. The actual energy release rate would depend upon the amount of fuel at any given temperature. It is very reasonable to assume that only a small fraction of the core would attain temperatures greater than 1600°C, because the emergency fire-water system would keep the core covered with water. This water would absorb the energy released, because the time scale is long (seconds rather than milliseconds).

Although an ignition type of rapid metal-water reaction could occur, the probability of obtaining the required temperatures is considered quite low because the core tank will be full of water. Also, between 1600 and 1750°C, several minutes delay in ignition may exist and thus provide a greater period of time to produce quenching of the fuel since reactor power is decaying. It is also highly improbable that a 100 percent reaction of this nature would occur because the reactor fuel would not all be at this high a temperature. However, the vessel would normally contain enough water to more than adequately absorb the energy released.

The 100 percent reaction would produce some 56 pounds of hydrogen gas which could be liberated to the reactor building in the event of a primary system rupture. Assuming that the hydrogen will disperse throughout the gastight volume

FIG. XVI-3-3 ALUMINUM-STEAM REACTIONS, LEVITATION METHOD (8 MM. SPHERE).

of 1,640,000 ft³, the hydrogen concentration would only be 0.86 percent, which is a factor of 4.8 below flammability limit and 22 below the explosive limit. Although higher concentrations are possible, the factors above, with respect to concentration and the need for the very high fuel temperature to obtain these release rates associated with ignition, lend confidence to a prediction that a hydrogen explosion will not occur.

4. FUEL MELTING DURING A FLOW BLOCKAGE ACCIDENT

The analysis of fuel melting accidents caused by restriction or blockage of the fuel element coolant channels would be similar to that presented in the previous section on a loss of primary coolant accident. However, there are three major differences in the accident. There will be several seconds delay between the start of melting and a reduction in reactor power since the mode of detection is a fission break monitor system located in the reactor discharge piping. The primary system is intact, and the fission products will not be released to the building. The amount of fuel involved will be much less than the entire core and could be restricted to two plates.

A pessimistic assumption would be to base this analysis on the energy release from the entire core as was done for the rupture accident. On this basis, both the thermite and metal-water energy releases would be essentially the same as for a loss of coolant accident. The reactions are slow enough that shock waves would not be generated, and the energy could be transferred to the reactor coolant. The hydrogen released would be contained in the primary system and would collect in the top of the reactor vessel. This could be safely vented or slowly removed by operation of the system degassing system. The maximum amount of hydrogen generated (56 pounds) would require a volume of approximately 440 cubic feet at reactor pressure. This would probably cause the safety valves to lift, preventing large overpressurization. The capacity of both safety valves is 1200 gallons of water per minute. This blowdown capacity is sufficient to handle the volume of water displaced by hydrogen generated by 18 percent of the core reacting at a maximum rate of 10 percent per second (this is the reaction rate of a 5 mm sphere).

5. REACTOR VESSEL INTEGRITY

5.1 General

The capacity of the reactor vessel to absorb the energy of a nuclear excursion without suffering a major fracture is dependent on a number of factors. A water-filled vessel made of a tough ductile material such as stainless steel is capable of containing large energy releases without failure. The ability to absorb the energy release is largely a function of the time scale of the release. Fast explosive-type shocks are more destructive than pressure rises over a few msec, even for identical energy releases. Experimental studies of the problem have been made by Stanford Research Institute and the Naval Ordnance Laboratory. However, most of the available data is in the range of explosive shocks and leads to results which are pessimistic when compared to the slower pressure rises encountered in nuclear excursions.

5.2 Slow Pressure Rise

The time required to obtain peak pressure in the reactor vessel is estimated to be at least equal to the reactor period. This conclusion is based on examination of transient pressure data [1], which showed variations in pressure rise time ranging from over twice the reactor period to slightly above one reactor period.

Assuming a pressure rise of several msec duration and propagating at sonic velocities, the pressure throughout the reactor vessel at any moment may reasonably be assumed to be uniform for purposes of calculating the total energy capacity.

The analytical model assumed a stainless steel cylinder 11-foot, 8-inch ID by 2 inches thick by 35 feet high containing 64 cubic feet of solid material inside the vessel and 3680 cubic feet of water. These dimensions are those of the ATR vessel neglecting the effects of nozzles and flanges.

The basic assumptions for the solution for a slow-pressure rise condition are:

- (1) All energy is released in the center of the vessel and is in the form of steam.
- (2) As the energy is released over several msec, it is free to expand outward doing work of compression and deformation. The steam bubble is assumed to have an adiabatic boundary so that no heat is transferred. This steam bubble may be thought of as a piston of expanding dimensions; the water surrounding the bubble is compressed. As the pressure on the wall increases, the vessel deforms and work of deformation occurs.
- (3) At the end of the energy release, equilibrium pressure is reached, and the solution can be treated as a static energy summation.

Calculations were made for this model over a range of final pressures from 400 to 1600 psig. Each calculation assumed a final pressure condition, and the total energy was calculated to be the sum of the energy remaining in the bubble of steam, the energy absorbed by isentropic compression of the water, and the energy absorbed in deforming the vessel. Results are shown in Figure XVI-3-4 for an unrestrained vessel and one restrained by the concrete shielding. The actual values lies between the two curves.

In order to help establish a lower limit of the energy released, a series of conservative assumptions was made:

- (1) The pressure wave is not uniform throughout the reactor vessel, but is uniform in a spherical shape equal to the diameter of the vessel.
- (2) Only a portion of the total water is compressed, and a smaller part of the reactor vessel increases in volume.

FIG. XVI-3-4 EFFECT OF ENERGY RELEASE ON CYLINDRICAL VESSEL.

The resultant equilibrium pressure is higher for a given total energy release. This relationship is shown in Figure XVI-3-5. The energy level at which the vessel wall comes into contact with concrete shielding is about 225 MW-sec.

The reactor vessel is designed for a pressure of 390 psig. The rupture pressure based on the initial dimensions and the ultimate strength of the material is 1890 psig. The total fast neutron exposure of the reactor vessel is less

FIG. XVI-3-5 EFFECT OF ENERGY RELEASE ON SPHERICAL VESSEL.

than 10^{16} n/cm² (>0.4 eV) for a 20-year life. Therefore, essentially no change in properties of the stainless steel vessel will occur over the assumed life of the reactor.

Analytical and experimental static rupture tests [12, 13] show that the actual rupture pressure of idealized vessels is not less than 90 percent of the ultimate strength and that large deformation can take place when ductile material is used.

In the preceding analysis it was estimated that a simple vessel similar in size to the ATR vessel will rupture at 1890 psig. However, this value must be corrected to include the available rupture test data and to account for the penetrations in the actual vessel. These considerations lead to the conclusion that the ATR vessel will not rupture at a pressure less than 1200 psig as a result of a nuclear excursion. A pressure of 1200 psig corresponds to a nuclear energy release of 1200 MW-sec in an unrestrained cylindrical vessel or 350 MW-sec in a fully restrained cylindrical vessel as shown in Figure XVI-3-5.

It may be concluded that for the maximum nuclear energy release of 130 MW-sec, which is a factor of 2.7 to 9 below the above rupture values, the ATR vessel will not rupture.

5.3 Explosive Pressure Rise

In order to provide a direct correlation between reactor excursion energy releases and the consequent damage, a number of model tests has been conducted by Stanford Research Institute and the Naval Ordnance Laboratory. Most of these tests have been based on the assumption that a TNT or equivalent explosion is similar to an actual reactor energy release. The high velocity shock waves and large gaseous product bubble generated by this class of explosive make this assumption conservative, but it does provide a direct relationship between energy release level and vessel capability.

The nuclear energy release is divided between the heating of the fuel plates and water, and the formation of steam. When considering rapid pressure rises, no more than half of the energy occurs in the steam. Therefore, when considering an equivalence between explosive energy release and nuclear energy release, a ratio of 2 should be used.

The basis for using model experiments to establish the pressure deformation and damage of explosive energy releases has been reported experimentally and analytically by Baker [14] and Huber and Zabel [15].

Model experiments attempting to simulate reactor excursions, wherein the vessel was full of water and did not fail, provide two points predicting the lower limit at which this vessel may fail.

- (1) Calculations, supported by previous model tests, by Dr. W. R. Wise, Jr., of the Naval Ordnance Laboratory show that the ATR vessel (idealized by assuming no nozzles) will withstand the shock wave and blast effects of a TNT explosion (of microseconds

duration and 100 percent gaseous products) of at least 125 pounds. The equivalent nuclear energy is 236 MW-sec. For energy release rates in the order of several msec, the containment potential would be considerably greater.

- (2) Experiments by the Stanford Research Institute [15] show that a simple vessel of similar size can withstand 1500 MW-sec without rupture if the explosive periods are prolonged to 1 msec. By proportioning this energy release to the ATR vessel dimensions, an extrapolation of the ATR capability can be made.

The SRIA test model was a 4-1/4-inch OD cylinder with 0.120-inch wall of SAE 1020 carbon steel at 1/24 scale. This is equivalent to a full-size vessel of 102-inch OD with 2.88-inch wall thickness. By assuming 45,000-psi ultimate tensile strength for this material, the energy absorption capacity of the vessel material in the core region, assuming vessel height equal to diameter, is proportional to:

$$WK_v = V_{\text{metal}} \times UTS = 83,000 \text{ in.}^3 \times 45,000 \text{ psi} = 3.735 \times 10^9 \text{ lb-in.}$$

For explosive energy release rates, only the vessel material near the core will be subjected to the maximum shock wave so that the assumption as to vessel height is reasonable. The equivalent value of energy absorption for the ATR vessel is:

$$WK_v = 123,000 \text{ in.}^3 \times 70,000 \text{ psi} = 8.61 \times 10^9 \text{ lb-in.}$$

The ratio of energy capability is:

$$\text{ATR/SRIA test} = \frac{8.61 \times 10^9 \text{ lb-in}}{3.735 \times 10^9 \text{ lb-in}} = 2.3.$$

Since the energy is dissipated in the water as well as the vessel wall, the relative volume of water in one diameter vessel height for each vessel is also proportional to its energy absorption capacity.

$$\text{Water volume ratio} = \text{ATR/SRIA test} \frac{2.15 \times 10^6 \text{ in.}^3}{0.69 \times 10^6 \text{ in.}^3} = 3.1.$$

Using the lesser ratio of capacity, the ATR vessel should contain 1500 MW-sec $\times 2.3 = 3450$ MW-sec if no nozzle reinforcements were present. By allowing a generous reduction in the capability of the vessel for the weakening effect of nozzles, it can be concluded that the actual ATR vessel would be capable of containing at least 1500 MW-sec.

The simulation of reactor excursion energy by TNT explosions or by slower burning pellets is not very satisfactory because of the differences between them and the reactor excursion. TNT or Pentolite explosions occur in microseconds, produce ultrasonic shock waves, produce large gas bubbles, and create more damage than is expected from an energy release produced at a slower rate. Estimates based on the BORAX I, and the model test work at independent explosive research laboratories, all show that the damage from explosives is 3 to 10 times greater than is produced by the same energy release in nuclear reactors, or by tests at a duration of 1 msec or longer [1, 2].

5.4 Summary

Based on the analysis presented in this section, it is concluded that the ATR vessel will not rupture as a result of the maximum credible energy release. Analyses have been made using three different approaches:

- (1) Assuming the pressure rise is slow enough to assume uniform pressure throughout the vessel, calculated pressure rises have been evaluated.
- (2) Using experimental data, slow pressure rises have been evaluated.
- (3) Using experimental data, rapid pressure rises (time intervals of a few microseconds) have been evaluated.

It has been shown in the Spert testing that no more than one-half the energy released in the nuclear excursion is rapidly transferred to the coolant. If the nuclear excursion produces steam, only one-half of the total energy could be considered available as pressure energy. Thus the safety factors on the vessel capacity discussed below may be increased by a factor of 2.

When the pressure rise is slow enough so that the pressure differential throughout the reactor vessel is small, the reactor vessel can accommodate the MW-sec energy release associated with the maximum reactivity addition without rupture. The release would cause a peak pressure of 850 psig. Energy releases of 2.7 to 9 times the above maximum energy release of the nuclear accident would be required to rupture the vessel.

Based on experimental work performed by the Stanford Research Institute, it is concluded that the ATR vessel should be capable of containing at least 1500 MW-sec from an energy release with a period on the order of 1 msec. This provides a safety factor of >10 on the 130 MW-sec resulting from the maximum reactivity addition.

Work by Dr. W. R. Wise, Jr., has shown that the ATR vessel (idealized by assuming no nozzles) can withstand a TNT explosion of 125 pounds or 236 MW-sec of energy release. To allow for the difference in time scale between reactor energy releases and TNT explosions, this capability can be increased by a factor of at least 3. The capability of the ATR vessel based on this work is 708 MW-sec. This is a factor of 5.4 times the 130 MW-sec energy release resulting from the maximum energy release of the nuclear excursion.

The most stringent of the above types of energy releases is the TNT explosion. Using the above data from Dr. Wise's work, it is concluded that the ATR vessel will withstand the detonation of at least 125 pounds of TNT which is equivalent to 236 MW-sec.

XVI-4 REFERENCES

1. J. C. Haire, Subcooled Transient Tests in the Spert-I Reactor -- Experimental Data, IDO-16342 (July 1958).
2. J. R. Dietrich, Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled, Water-Moderated Reactor, AECD-3668 (1954).
3. W. E. Baker and J. D. Patterson, Blast Effects Tests of a One-Quarter Scale Model of the Air Force Nuclear Engineering Test Reactor, BRL-1011 (March 1957).
4. R. O. Brittan, Reactor Containment (Including a Technical Progress Review), ANL-5948 (May 1959).
5. R. G. Wheeler and D. C. Kaulitz, Investigation of Fracture in High Pressure Loop Through CG 6 x 9 Position of the ETR, Hanford Atomic Products, HW-63556 (January 18, 1960).
6. L. F. Epstein, "Correlation and Prediction of Explosive Metal-Water Reaction Temperatures", Nucl. Sci. Eng., 10 (July 1961) pp 247-53.
7. Chemical Reactions, SIFTOR, Chapter 16 (to be issued).
8. R. W. Miller et al, "Experimental Results and Damage Effects of Destructive Test", Trans. Am. Nucl. Soc., 6 (June 1963) pp 138-39.
9. General Electric Company, Final Report of SL-1 Recovery Operation, May 1961 Thru July 1962, IDO-19311 (July 1962).
10. Argonne National Laboratory, Reactor Development Program Progress Report, May 1963, ANL-6739 (June 1963).
11. R. E. Wilson, "Kinetics of the Molten-Aluminum/Steam Reaction by the Levitation Method", Trans. Am. Nucl. Soc., 6 (June 1963) p 102.
12. W. E. Cooper, "The Significance of the Tensile Test to Pressure Vessel Design", Welding J., 36 (June 1957) pp 49s-54s.
13. J. Marin and M. G. Sharma, "Design of a Thin-Walled Cylindrical Pressure Vessel Considering Plasticity", Welding J., 37 (1958) pp 515s-519s.
14. W. E. Baker et al, Laws for Large Elastic Response and Permanent Deformation of Model Structures Subjected to Blast Loading, BRL-1060 (December 1958).
15. G. B. Huber and N. R. Zabel, Containment of Fragments From a Runaway Reactor, SRIA-36 (October 1960).

XVII. HAZARDS ANALYSIS

- XVII-1 Minor Radiological Hazards
- XVII-2 Environmental Hazards Associated with Normal Operation
- XVII-3 Environmental Hazards Associated with Large Releases of Radioactivity
- XVII-4 Maximum Hypothetical Accident
- XVII-5 The Consequences of an ATR Accident
- XVII-6 References

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

XVII-1. MINOR RADIOLOGICAL HAZARDS

1. GENERAL

In the operation of a test reactor there are a number of minor incidents which constitute a radiological hazard within the plant, yet do not present a significant environmental hazard. Dependence must be placed on the radiation monitoring equipment and administrative procedures for the control of these hazards and the protection of personnel. These accidents can be grouped as follows:

- (1) Fission breaks into the primary coolant
- (2) Fission breaks into the experiment loop coolant
- (3) Releases of radioactive gases into the building air
- (4) Releases of solid or liquid radioactive materials into the building.

2. FISSION BREAKS INTO THE PRIMARY COOLANT

During the operation of the reactor, there may be occasional periods when the primary coolant will be contaminated with fission products from defective fuel elements or experiments. The resulting radiological hazard is limited by the integrity of the primary system and its shielding. The ATR shielding will limit the radiation level increase to 5 mrem per hour, for a release of all the fission product inventory of one fuel element (after 17 days' operation) uniformly distributed in the primary system.

When the primary system becomes contaminated, the Fission Break Monitor will detect the increased activity. Additional indication of the radiation level is provided by a monitor on the degassing tank exhaust line. Alarm signals from the radiation monitoring system will alert the operator to take the necessary steps to prevent excessive exposure of personnel.

If the contamination is from small fission breaks, the radiation level outside the shielding will not increase significantly; however, shutting down the reactor will in any event reduce the amount of activity released.

The contamination can be removed from the primary system by demineralizing, controlled degassing, and then by flushing prior to removing the damaged elements or experiments.

3. FISSION BREAKS INTO EXPERIMENT LOOP COOLANT

Experimental programs that are intended to develop improved fuel materials often require operation up to or beyond specimen failure. The failure

of a specimen will contaminate the test loop coolant with fission products and require the safe disposal of radioactive liquids and gases.

The shielding around the pressurized water loop cubicles will limit the radiation level outside the cubicle after an experiment failure. A specimen failure (previously described) results in a one-hour integrated dose outside the cubicle of 12 mrem.

The pressurized water loops have integral ion exchange units for cleaning up contaminated coolant. The loops may be drained, vented, and flushed through a closed system to shielded waste collection tanks. The loops may be chemically decontaminated if necessary.

The gas-cooled loop, which has not yet been designed, will be provided with a suitable gas cleanup, holdup, and vent system as required for the disposal of the waste gas from the closed loop.

4. RELEASE OF RADIOACTIVE GAS INTO THE BUILDING AIR

The release of activity into the ATR building air from normal operation is limited by maintaining the vent and drain systems under continuous suction and design of air flow patterns to prevent backflow of contamination. At MTR and ETR, minor releases of radioactive gases into the building air have occurred in the operating history and must be considered for ATR. Radiolytic gas releases have occurred in the MTR and ETR. Although some of these releases have interfered with operation, they did not constitute a major hazard. In a number of cases, the evacuation of the entire reactor building has been required, but usually only parts of the building required isolation. In a very few cases, the atmospheric conditions permitted contaminated stack discharge air from one reactor to enter the neighboring reactor building and required its evacuation.

The radiation monitoring equipment in the ATR building will detect air contamination and sound an alarm. The radiation monitoring system will also isolate certain building areas by shutting off ventilation. If the radiation or air contamination levels exceed safe limits, the reactor will be scrammed and the planned ATR evacuation procedure instituted.

5. RELEASE OF SOLID OR LIQUID RADIOACTIVE MATERIALS INTO THE BUILDING

Spillage accidents at MTR and ETR usually involve minor leakage of reactor or experiment coolants. While not usually a serious radiation hazard in themselves, these spillages may contaminate the building air. Under such conditions monitors will sound an alarm and appropriate safety measures can be taken. The reactor cooling system and the experiment cooling systems are designed, fabricated, and tested to assure maximum integrity. Consequently, spillage accidents are expected to be only minor nuisances. Furthermore, all floor areas include ample drains connected to the waste tanks which provide for spillage runoff and permit immediate washdown. The waste tanks and the drains are vented to the waste gas system; this provides a continuous flow of sweep air for positive removal of airborne activity.

XVII-2. ENVIRONMENTAL HAZARDS ASSOCIATED WITH NORMAL OPERATION

During normal operation the ATR will produce radioactive gaseous effluents which are vented to the stack. These effluents are variable in concentration and quantity. A review of the operating history of ETR is used to evaluate the acceptability of the ATR effluent.

During 1959, when a large open cycle air-cooled experiment was operating in ETR, its stack gas effluent carried up to 5 curies per megawatt-day or a total of 875 curies per day. An open cycle air-cooled experiment is not planned for ATR. The calculated normal gas effluent at ATR is expected to be 40 curies per day of Kr and Xe. The Argon-41 activity produced from shield cooling is negligible.

Air monitor readings in the ETR building are correlated with releases of activity from the ETR stack in IDO-16623 (Reference 1). The data indicate that an ETR stack release of 875 curies per day would result in a maximum specific activity of 1.6×10^{-9} curies per cubic meter in the ETR building. The records indicate that this specific activity was exceeded only 2.4 percent of the time that the stack discharge averaged 875 curies per day. The corresponding value of the maximum specific activity during normal operation of the ATR is expected to be only 0.075×10^{-9} curies per cubic meter. In view of the more favorable location of the ATR stack with respect to the three reactor buildings, the expected maximum specific activity and its frequency of occurrence in any of the three buildings as a result of ATR operation should be substantially less than the levels produced by the ETR.

In any event, the expected level is significantly less than 4×10^{-9} curies per cubic meter permitted in National Bureau of Standards Handbook No. 69 (Reference 2) for a 40-hour week continuous exposure to mixed beta-gamma activity, and a factor of 10^{-4} below the permissible specific activity for the noble gases. Hence, there will be no significant environmental hazard from normal operation of ATR.

XVII-3. ENVIRONMENTAL HAZARDS ASSOCIATED WITH LARGE RELEASES OF RADIOACTIVITY

1. GENERAL

The environmental hazard associated with accidents leading to large releases of radioactivity has been analyzed. The results are presented in this section. All computations are based on building leakage at the rate of 10 percent of the building volume per day.

The fission product release assumed for the calculations is based upon a major accident hypothesized for purposes of site analysis and postulated from consideration of possible accidental events that would result in potential hazards not exceeded by those from any accident considered credible.

2. FUEL TRANSFER ACCIDENT

2.1 General

An operational accident can result in the release of the fission products in any single fueled component. The largest fueled component is a single fuel element which contains three percent of the total core fission product inventory. In the analysis of the fuel transfer accident, it is assumed that the released fission products are distributed immediately throughout the air in the gas-tight volume of the building. The environmental hazard results from direct radiation from the contaminated air in the superstructure and the leakage of the air from the building. Although short term leak rates greater than 10 percent per day occur, dose rates to off-site areas are based on long term average values. Dose rates to TRA personnel are based on short term adverse weather that will not result in an average leak rate of 10 percent per day. The unknown factor of time required to disperse fission products uniformly in the superstructure also will reduce short term fission product release rate. Based on these factors, an average leak rate of 10 percent per day was used in the calculations.

In melting one fuel element, the fission products released are three percent of the quantities shown in 10 CFR-100 (Reference 3). This accident releases the following percentages of the core fission product inventory after 17 days of operation at 250 MW:

3.0 percent of the rare gases

1.5 percent of the halogens, half of which are assumed to be absorbed on building and equipment

0.03 percent of the solid fission products.

The core fission product inventory after 17 days of operation at 250 MW is shown in Appendix C.

During the course of the fission product release, the fission products in the building are assumed to undergo radioactive decay. No credit is taken for decay in the plume en route, for fallout, nor for rainout. These assumptions are in accordance with the recommendations in 10 CFR-100:

The results calculated for this accident are believed to be conservative for the following reasons:

- (1) Fuel melting will not occur instantaneously
- (2) The fission products will not be distributed immediately throughout the superstructure
- (3) The fission product inventory is conservatively high since a minimum of five hours decay after shutdown is required before transferring fuel.

3. MAXIMUM HYPOTHETICAL ACCIDENT

3.1 General

A complete meltdown of the ATR core has been defined as the maximum hypothetical accident.

For the purposes of analysis, it is postulated that the ATR core suffers a total meltdown, and that the fission gases are released to the pipe tunnel in the basement. At the time of the accident it is assumed that the 12- x 12-foot machinery hatches, in both the operating floor and the first basement floor, are open and a 10°F temperature differential exists; the contaminated basement air mixes with the air in the superstructure by natural convection. It is assumed that the entire release is uniformly distributed in the gastight building volume when the volume of air in the superstructure has been displaced by an equal volume from the basement.

The release of fission products associated with complete melting of the core is taken as equal to the release shown in 10 CFR-100 (Reference 3). Accordingly, the following percentages of the core fission product inventory after 17 days of operation at 250 MW are released to the building air:

100 percent of the noble gases

50 percent of the halogens, half of which become absorbed on building and equipment surfaces

1 percent of the solid fission products.

The core fission product inventory after 17 days of operation at 250 MW is shown in Appendix C.

The ATR building spray system, although not considered an engineered safeguard, can result in very appreciable reductions in iodine concentration in the building air [7, 8]. Calculations of dose rates with sprays were made for

reference accidents assuming a reduction of iodine concentration in building air to one percent of the initial concentration in 10 hours. The spray system can also reduce solid fission product concentration and the reference calculations for the effect of the sprays assumed a reduction to 25 percent of the initial concentration of solid material. The sprays have no effect on the noble gas fission products.

3.2 Time Scale of Accident

The time required to melt the fuel and to release the fission gases into the building superstructure is important because, once in the superstructure, the fission products create an environmental hazard from the direct radiation and from the leakage of contamination from the building.

The accident can be considered as occurring in the following sequence:

- (1) The melting of the core occurs first. This phase is of short duration and is considered as occurring instantaneously.
- (2) The fission gases then escape from the reactor vessel into the pipe tunnels by thermal convection. If it is assumed that the reactor vessel has two openings 24 inches in diameter, 20 feet apart in elevation, and temperatures in the vessel and in the pipe tunnel are 212 and 130°F, respectively, then the vessel air volume of 2860 cubic feet will be exchanged with the tunnel in approximately one minute.
- (3) The gases are then distributed by thermal convection from the pipe tunnel to the building basement. The most probable path for the gases is through various small openings in the walls of the cubicles. The cubicles are assumed to have the equivalent of a six-inch diameter convection path communicating with the pipe tunnel and a similar path communicating with the basement. These convection paths are maintained by an anticipated temperature differential of 50°F and a difference in elevation of 20 feet. Under these conditions, the transfer of 2860 cubic feet of gas will take seven minutes.
- (4) The contaminated basement air then mixes by thermal convection with the remainder of the confinement volume through the 12-foot-square equipment hatches, between the main floor and basement levels. With convection currents maintained by a 10°F temperature difference and a 40-foot difference in elevation, the time to displace the superstructure volume is 24 minutes.

The time sequence of this postulated series of events is shown in Table XVII-3-1.

3.3 Conservatism in Analyses

The results of the postulated maximum hypothetical accident calculations are conservative for the following reasons:

- (1) The function of the hatches is such that they would be expected to be open primarily when the reactor is shut down. Furthermore,

TABLE XVII-3-1

TIMETABLE FOR POSTULATED COMPLETE CORE MELTDOWN

Time -- zero

Reactor is operating at full power.
Machinery hatches are open.
Building ventilation is normal.
Accident occurs.
Evacuation alarm sounds, almost immediately.

Time -- 1 minute

Ventilation system is shut off, and the gastight volume is sealed.
Building leakage starts at rate of 10% of building volume in 24 hours (see Section 2.1).
Fuel elements have melted.

Time -- 2 minutes

Evacuation of MTR-ETR-ATR site begins.
Building sprays are started.

Time -- 8 minutes

All of the core has melted, and 100% of the fission product release has entered the basement from the pipe tunnel and is uniformly mixed in basement.
Diffusion into the superstructure starts at a linear rate.

Time -- 15 minutes

Buses start to arrive at evacuation area and begin removing personnel.

Time -- 30 minutes

Last bus has left, and the evacuation of the MTR-ETR-ATR site is complete.

Time -- 32 minutes

The released fission products are uniformly distributed through the entire confinement volume of the ATR building.

they are not likely to be open at the time of the accident because they must be kept closed to avoid disturbances of the ventilation pattern.

(2) The postulated 10°F temperature difference between basement areas and main floor is that which is normally maintained during operation of the ventilation system and is greater than that which could be maintained with the hatches open.

(3) The entire fission product release will not be uniformly mixed into the building air when one superstructure volume has been displaced; rather the concentration of fission gases in the basement will be much greater than in the superstructure.

(4) A complete core meltdown represents an improbable event because of the core tank design and emergency fire-water system.

(5) The spray system, although not credited in the maximum dose calculations, can reduce the concentration of halogen and solid fission products.

(6) No decay of cloud is considered after it leaves the building.

(7) The core meltdown must occur near the end-of-fuel-cycle further reducing the postulated accident probability.

XVII-4. DOSE CALCULATIONS

1. GENERAL

The dose calculations used for the fuel handling accident and maximum hypothetical accident are standard. Details and equations used in the calculations are shown in Appendix B.

2. METEOROLOGICAL FACTORS

In order to evaluate the environmental hazard from an ATR accident, the integrated dose and instantaneous dose rates were calculated using the Sutton equation and the conversion constants given in TID-14844 (Reference 9) and AECU-3066 (Reference 4). The Sutton equation relates the diffusion of gases to atmospheric conditions.

$$\frac{X}{Q} = \frac{2}{\pi u C_y C_z d^{(2-n)}}$$

where

X = concentration at a point d meters downwind from the source
(curies per cubic meter)

Q = release rate (curies per second)

u = average wind velocity (meters per second)

C_y and C_z = diffusion coefficients in the horizontal and vertical
directions, in planes perpendicular to the d direction

n = atmospheric stability constant.

The atmospheric dilution factor, $\frac{X}{Q}$, which is defined by Sutton's equation, must be evaluated for one selected set of meteorological parameters. These parameters as used in the calculation are invariant with time. Although the selected parameters must be conservative, the calculated value of the atmospheric dilution factor must be a realistic estimate of the dilution factor which can actually be experienced during the most adverse weather conditions. In order to more accurately account for the invariant nature of the weather conditions in the calculations, the following two sets of NRTS weather conditions have been used:

- (1) Short term adverse weather conditions
- (2) Long term adverse weather conditions.

2.1 Short Term Adverse Weather

Short term adverse weather conditions are of interest to determine (a) the maximum doses received during an evacuation, (b) the maximum dose rates at various NRTS facilities during the period following the accident, and (c) accessibility to evacuated facilities.

Because unfavorable weather can exist simultaneously with the accident and can occur when reentry to a facility is planned, the values referred to in (1) and (2) above were determined using the most unfavorable weather conditions of record at WBO (IDO-12015, Reference 5) which are as follows:

(1) A persistent inversion, where $n = 0.5$. This value coupled with the appropriate values of C_y and C_z gives a conservative estimate of diffusion.

(2) An invariant wind, where $u = 1.23$ meters per second. This wind velocity was determined by averaging the lowest reported velocity winds, from 0 to 2.45 meters per second.

(3) A C_z constant corresponding to the reported n and u values. A C_y constant using the relationship $C_y = C_z$.

These meteorological parameters are hereafter referred to as Case A weather conditions.

2.2 Long Term Adverse Weather

Long term adverse weather is of interest to determine the maximum integrated doses for various locations surrounding the ATR, up to and including the largest population center within 50 miles. The value of the maximum integrated dose determines whether evacuation is required following an ATR accident. All integrated doses have been calculated for the time span from the time of the accident to infinity.

One criteria for long term adverse weather is established in TID-14844 (Reference 9). The conditions defined in TID-14844 are representative of poor conditions throughout the United States and not necessarily truly representative of NRTS conditions. However, this criteria, where $n = 0.5$, $u = 1$ meter per second, $C_y = 0.4$, and $C_z = 0.07$, has been used as a reference point for comparison with the results based on the long term adverse weather conditions. This 10 CFR-100 case is hereafter referred to as Case I.

Since a large amount of weather data is available for the NRTS, a detailed study was conducted to determine the worst long term conditions that could be expected over the time scale of the accident. Each 30 day period was studied to establish the month in which the inversions most severely reduced the diffusion.

The diffusion conditions for the worst month and the worst winter month at NRTS, hereafter referred to as Cases II and III, respectively, were determined as follows:

(1) By examining Figure II-1-1, it was determined that the month of September exhibits the most adverse weather conditions because it has the severest inversion conditions for the most hours per

day. Although January has more total hours of inversion, it was found that the lesser severity of the inversions resulted in more favorable conditions.

(2) The time weighted average wind velocities (independent of direction) for each reported atmospheric stability constant for September and January were determined from the appropriate wind roses in Figures II-1-7 through -12.

(3) The values of the diffusion coefficient C_z were selected from IDO-12015 (Reference 5), page 140, for the appropriate wind velocities and atmospheric stability conditions. The values of C_y were calculated using the following relationships which are valid for NRTS and are presented in IDO-12015 (Figures 48-53):

	$\frac{C_y C_z}{}$
Strong Inversion ($n = 0.50$)	8
Mild Inversion ($n = 0.33$)	4
Mild Lapse ($n = 0.25$)	1
Strong Lapse ($n = 0.20$)	1

(4) The value of $\frac{X}{Q}$ at several distances was calculated for each of the four stability conditions using Sutton's equation.

(5) These values of $\frac{X}{Q}$ calculated above were weighted by the average number of hours per day that each of the four stability conditions prevail during the respective months of September and January, as determined from Figure II-1-5. This results in a weighted average value of $\frac{X}{Q}$ as a function of distance for the month with the most adverse weather conditions and for the worst winter month.

Figure XVII-4-1 shows $\frac{X}{Q}$ as a function of distance for:

- (a) The short term inversion, Case A
- (b) The 10 CFR-100 criteria, Case I
- (c) The worst NRTS month, September, Case II
- (d) The worst NRTS winter month, January, Case III.

Table XVII-4-1 summarizes the parameters used in the four basic cases.

The effect of the use of sprays on the integrated doses has been calculated for Case II, the worst NRTS month. The results of these calculations are reported as Case II-S. Similarly, the effect of sprays on the dose rates resulting from short term weather conditions, Case A, is reported as Case A-S.

FIG. XVII-4-1 ATMOSPHERIC DILUTION FACTORS BASED ON SUTTON'S EQUATION.

TABLE XVII-4-1

METEOROLOGICAL PARAMETERS

Atmospheric Stability Constant n	CASE I				CASE II				CASE III			
	Short-Term Adverse Weather				Long-Term Adverse Weather LOCFR-100				Long-Term Adverse Weather NRTS Worst Month (September)			
	Daily Duration (%)	Wind Velocity (m/sec)	C _z	C _y	Daily Duration (%)	Wind Velocity (m/sec)	C _z	C _y	Daily Duration (%)	Wind Velocity (m/sec)	C _z	C _y
0.50	100	1.23	0.036	0.288	100	1.0	0.07	0.40	0.40	2.9	0.029	0.232
0.33	0	--	--	--	0	--	--	--	0.13	2.9	0.069	0.276
0.25	0	--	--	--	0	--	--	--	0.07	5.2	0.139	0.139
0.20	0	--	--	--	0	--	--	--	0.40	5.2	0.299	0.299

The wind direction at NRTS generally rotates around the compass daily. The wind at NRTS, during the month of September (Case II), blows in one direction (22.5 degree sector) for less than 22 percent of the time. This frequency is based on records for approximately 12 years. The effect of this changing wind direction on integrated dose for the worst NRTS month is reported here as Case II-W.

The cases, when treated individually, indicate the upper bound of the environmental hazard. An additional case was analyzed to indicate a more reasonably expected condition over the time scale of the accident; this case is referred to as Case II-S-W and is based on the effect of building sprays and the variation of wind direction on the doses calculated for the worst long term NRTS weather condition (Case II).

3. CONSERVATISM IN ANALYSIS

There are other factors which act to reduce the environmental hazard calculated under either of the four cases. The most significant of these factors which can realistically be expected to influence the actual integrated doses and dose rates are:

(1) Building sprays

(2) No credit was taken for cloud meandering. The following correction factors are derived from meandering over 22-1/2° sector obtained from AECU-3066 (Reference 10):

<u>Distance -- Meters</u>	<u>Strong Lapse</u>	<u>Strong Inversion</u>
10^2	0.95	0.40
10^3	0.75	0.22
10^4	0.60	0.13
10^5	0.47	0.07

(3) No credit was taken for depletion of cloud, via deposition, en route to receptor. This has been estimated (1.0 cm/sec iodine, 0.2 cm/sec solids as deposition velocities) as decreasing doses at 10^4 meters by a factor of 5 for iodine.

(4) The occurrence of the most adverse weather conditions used in the analyses is not compatible with the maximum possible leak rate. Case A weather, $U = 1.23$, is coupled with a wind velocity of 11.7 miles/hour, $U = 4$, for the high building leak rate calculations.

(5) No credit was taken for cloud decay en route.

(6) Sutton's diffusion equation gives conservative estimates of diffusion in that it assumes the surface over which the cloud travels is aerodynamically smooth. In actual practice, it is almost invariably aerodynamically rough.

(7) The probability of the simultaneous occurrence of a complete core meltdown at the end of a fuel cycle, with the worst combination of leak rate and diffusion conditions, is extremely small.

(8) No record of wind direction persisting during stable atmospheres for more than 24 hours has been observed at the NRTS.

4. DOSE RATES AT MTR-ETR-ATR AREA

4.1 General

Points within this area are close enough to the ATR to receive an appreciable direct gamma dose from the gas cloud in the ATR building. However, the immersion beta-gamma dose rates and the iodine ingestion dose rates govern the evacuation time. These dose rates are computed for the short term adverse weather conditions, Case A.

4.2 Fuel Handling Accident

4.21 Dose Rates. The direct gamma dose rates at the MTR, the ETR, and the evacuation area are shown for this accident in Figures XVII-4-2, -3, and -4. In these figures, no credit is taken for the shielding effect of building structures.

The corresponding beta-gamma immersion dose rates at the same locations are shown for this accident in Figures XVII-4-5 and -6 and the iodine ingestion dose rates are shown in Figures XVII-4-7 and -8. These graphs also show the dose rate reductions that are possible by the use of building sprays.

These dose rates are used to determine the integrated doses received by personnel during the evacuation.

FIG. XVII-4-2 DIRECT GAMMA DOSE RATE AT THE MTR FOLLOWING A FUEL TRANSFER ACCIDENT.

FIG. XVII-4-3 DIRECT GAMMA DOSE RATE AT THE ETR FOLLOWING A FUEL TRANSFER ACCIDENT.

FIG. XVII-4-4 DIRECT GAMMA DOSE RATE AT THE EVACUATION AREA FOLLOWING A FUEL TRANSFER ACCIDENT.

FIG. XVII-4-5 BETA-GAMMA IMMERSION DOSE RATES FROM THE PLOME IN THE MTR-ETR-ATR AREA FOLLOWING A FUEL TRANSFER ACCIDENT.

FIG. XVII-4-6 BETA-GAMMA IMMERSION DOSE RATES FROM THE PLOME IN THE MTR-ETR-ATR AREA FOLLOWING A FUEL TRANSFER ACCIDENT.

FIG. XVII-4-7 IODINE INGESTION DOSE RATES TO THE THYROID IN THE MTR-ETR-ATR AREA FOLLOWING A FUEL TRANSFER ACCIDENT.

FIG. XVII-4-8 IODINE INGESTION DOSE RATES TO THE THYROID IN THE MTR-ETR-ATR AREA FOLLOWING A FUEL TRANSFER ACCIDENT.

4.22 Integrated Dose Received During the Evacuation of the MTR-ETR-ATR Area. During the evacuation of the area, the integrated dose received by personnel is of greater importance than the instantaneous dose rates. In calculating the integrated dose, the evacuation procedures were considered. It is known from the accident at the Chemical Processing Plant, reported in IDO-10036 (Reference 6), that buses can evacuate all personnel within 30 minutes after the time of an accident. Because of the short time involved, the building sprays are not a factor in reducing the dose to the evacuated MTR-ETR-ATR area personnel.

As a basis for calculation, the dose received by the guard was chosen. He is stationed closer to ATR than any other person except those in the ATR building. During evacuation the guard is assumed to (a) leave the guard house two minutes after the accident, (b) run for five minutes to reach the evacuation area 1/2 mile from the ATR, and (c) leave on the last bus 30 minutes after the accident. The integrated doses the guard would receive based upon Case A, short term adverse weather conditions, are shown in Table XVII-4-2.

TABLE XVII-4-2

MAXIMUM INTEGRATED DOSE RECEIVED BY GUARD
DURING EVACUATION AFTER A FUEL TRANSFER ACCIDENT

Location	Direct Gamma	Immersion Beta-Gamma	Ingestion Iodine
At guardhouse (2 minutes)	0.94 rem	6.3 rem	141 rem
Travel to evacuation site (5 minutes)	0.18 rem	3.1 rem	81 rem
Waiting at evacuation site (23 minutes)	0.01 rem	3.5 rem	86 rem
Total	1.13 rem	12.9 rem	308 rem

The iodine doses received in this accident are 2.5 percent in excess of the generally accepted 300-rem limit in 10 CFR-100 (Reference 3). The beta-gamma dose is considerably less than the generally accepted 25-rem limit.

Under actual conditions, the probability of receiving the above doses during evacuation is small, principally because inversions of the severity used in Case A, corresponding to $n = 0.5$, occur only 15 percent of the time and the wind blows toward the evacuation area less than 5 percent of the time.

In calculating the integrated dose received during evacuation, no credit was taken for shielding provided by structures, travel time of the gas plume, nor for the increased turbulence and consequent dilution of the gas plume flowing over and around structures in its path.

4.3 Maximum Hypothetical Accident

4.31 Dose Rates. The dose rates for direct gamma, immersion beta-gamma, and iodine ingestion for selected locations in the MTR-ETR-ATR area are

presented as a function of time in Figures XVII-4-9 through -16. The meteorological conditions corresponding to Case A, short term adverse weather with an invariant wind, were used to calculate the immersion beta-gamma and iodine ingestion dose rates.

Dotted curves show dose rates with building sprays and correspond to Case A-S. The effect of the building sprays is small shortly after an accident, but a large reduction is noted after several hours.

These curves are used in determining integrated doses received during evacuation and to evaluate the earliest time of reentry to MTR and ETR after the accident as discussed in Section XVII-5-1. The curves do not indicate the time scale of the accident in that dose rates are zero until the fission products diffuse into the building superstructure (eight minutes).

FIG. XVII-4-9 DIRECT GAMMA DOSE RATE AT MTR FOLLOWING A COMPLETE CORE MELT-DOWN.

FIG. XVII-4-10 DIRECT GAMMA DOSE RATE AT ETR FOLLOWING A COMPLETE CORE MELT-DOWN.

FIG. XVII-4-11 DIRECT GAMMA DOSE RATE AT EVACUATION AREA FOLLOWING A COMPLETE CORE MELTDOWN.

FIG. XVII-4-12 DIRECT GAMMA DOSE RATE AT ONE MILE FROM ATR FOLLOWING A COMPLETE CORE MELTDOWN.

FIG. XVII-4-13 BETA-GAMMA IMMERSION DOSE RATES FROM THE PLUME IN THE MTR-ETR-ATR AREA FOLLOWING A COMPLETE CORE MELTDOWN.

FIG. XVII-4-14 BETA-GAMMA IMMERSION DOSE RATES FROM THE PLUME IN THE MTR-ETR-ATR AREA FOLLOWING A COMPLETE CORE MELTDOWN.

FIG. XVII-4-15 IODINE INGESTION DOSE RATES TO THE THYROID IN THE MTR-ETR-ATR AREA FOLLOWING A COMPLETE CORE MELTDOWN.

FIG. XVII-4-16 IODINE INGESTION DOSE RATES TO THE THYROID IN THE MTR-ETR-ATR AREA FOLLOWING A COMPLETE CORE MELTDOWN.

4.32 Integrated Dose Received During the Evacuation of MTR-ETR-ATR Area. Using the same analyses used for the fuel transfer accident, the integrated dose received by the guard during evacuation is summarized in Table XVII-4-3 based upon Case A conditions.

TABLE XVII-4-3

INTEGRATED DOSE RECEIVED BY THE GUARD DURING
EVACUATION FOLLOWING COMPLETE CORE MELTDOWN

<u>Location</u>	<u>Direct Gamma</u>	<u>Immersion Beta-Gamma</u>	<u>Ingestion Iodine</u>
At guardhouse (2 minutes)	0 rem	0 rem	0 rem
Travel to evacuation site (5 minutes)	0 rem	0 rem	0 rem
Waiting at evacuation site (23 minutes)	<u>0.13 rem</u>	<u>12.6 rem</u>	<u>318 rem</u>
Total	0.13 rem	12.6 rem	318 rem

The iodine dose received in this accident is six percent in excess of the generally accepted 300-rem reference value. The beta-gamma dose is considerably less than the generally accepted 25-rem value.

The complete meltdown accident releases more activity and presents a greater environmental hazard than the fuel transfer accident. However, the difference in the times required to disperse the activity in the building results in similar evacuation doses in the two accidents.

Under actual conditions, the probability of receiving the above doses during evacuation is small, principally because inversions of the severity used in Case A, corresponding to $n = 0.5$, occur only 15 percent of the time and the wind blows toward the evacuation area less than 5 percent of the time.

In calculating the integrated dose received during evacuation, no credit was taken for shielding provided by structures, nor for the increased turbulence and consequent dilution of the gas plume flowing over and around structures in its path.

5. DOSE RATES UP TO 15 MILES OUTSIDE MTR-ETR-ATR AREA

5.1 General

Outside the MTR-ETR-ATR area, the direct gamma dose decreases to an insignificant level because of the distance. However, the immersion beta-gamma and the iodine ingestion dose rates may be appreciable during the periods of unfavorable weather as assumed in Case A.

5.2 Fuel Handling Accident

5.21 Dose Rates. The immersion beta-gamma dose rates resulting from the fuel transfer accident are shown as a function of time at various distances (2, 5, and 15 miles) on Figure XVII-4-17. The corresponding iodine ingestion dose rates are shown on Figure XVII-4-18. These curves show both Case A and Case A-S. The pronounced effect of the building sprays may be noted.

FIG. XVII-4-17 BETA-GAMMA IMMERSION DOSE RATES FROM THE PLUME AT DISTANCES UP TO 15 MILES FROM THE ATR FOLLOWING A FUEL TRANSFER ACCIDENT.

FIG. XVII-4-18 IODINE INGESTION DOSE RATES TO THE THYROID AT DISTANCES UP TO 15 MILES FROM THE ATR FOLLOWING A FUEL TRANSFER ACCIDENT.

The iodine dose rates at these distances are controlling in establishing the environmental hazard (without the use of sprays) because in a given time period they can produce a larger fraction of the reference integrated dose values than the immersion beta-gamma dose rates.

Reference to Figure XVII-4-18 shows that, without sprays, the maximum iodine dose rate at a two-mile radius is approximately 25 rem per hour at the time of the accident; this would allow at least 12 hours for evacuation of any facilities in line with the assumed invariant wind. Longer periods are available for greater distances, or with use of building sprays, or for points not on the axis of the plume. Reference to Figure XVII-4-18 also indicates that at the five-mile radius, the maximum dose rate at the time of accident is approximately 6 rem per hour without sprays, and it is unlikely that a total dose of 300 rem would ever be accumulated if personnel were not evacuated.

5.3 Maximum Hypothetical Accident

At other sites on the NRTS, or at more distant locations, the direct gamma dose rate decreases to an insignificant level. However, under adverse weather conditions, the immersion beta-gamma and iodine dose rates from the fission product plume can be appreciable and can remain at high levels for extended periods of time.

Figures XVII-4-19 and -20 show the immersion beta-gamma dose rates and iodine ingestion dose rates for distance up to 15 miles from ATR as a function of time. These dose rates were calculated for the Case A short term adverse weather conditions. These figures also show dotted curves for Case A-S conditions. A few hours after the accident, the effect of building sprays is pronounced, and results in considerable reduction in both the iodine and immersion beta-gamma dose rates.

The iodine dose rates at these distances are controlling in establishing the environmental hazard without sprays because in a given time they can produce a greater fraction of the permitted integrated dose than the immersion beta-gamma dose rates.

Reference to Figure XVII-4-20 shows that, without sprays, the maximum iodine dose rate at a two-mile radius is approximately 850 rem per hour, at

the time of accident, and 550 rem per hour at the end of the first day. The corresponding value with the use of building sprays at the end of the first day is less than 6 rem per hour. Sprays reduce the iodine dose rate from 850 to less than 100 rem per hour within the first five hours after the accident. At the point on the NRTS boundary closest to ATR, the iodine dose rate at the time of accident, without sprays, is 100 rem per hour. At this radius, a point in line with the assumed Case A invariant wind must be evacuated in approximately six hours. Longer periods are available for greater distances or with use of building sprays or for points not on the axis of the plume.

6. INTEGRATED DOSES UP TO 50 MILES FROM MTR-ETR-ATR AREA

The immersion beta-gamma and iodine ingestion doses were integrated from the time of the maximum hypothetical accident to infinity for the long term weather conditions, as defined by Cases I, II, III, II-S, II-W, and II-S-W. The resulting integrated doses as a function of distance are shown in Figures XVII-4-21 and -22.

Reference to Figures XVII-4-21 and -22 shows that for the long term adverse weather conditions (Case II), no point beyond 31 miles will receive integrated doses in excess of either 24 rem of beta-gamma or 300 rem iodine. The use of building sprays (Case II-S) reduces this distance to eight miles.

The above results are considered extremely conservative since the effects of wind meandering, and building leak rate calculated using meteorological conditions that do not occur for Case II weather, are neglected.

FIG. XVII-4-21 INTEGRATED IODINE DOSE TO THE THYROID.

FIG. XVII-4-22 INTEGRATED BETA-GAMMA IMMERSION DOSE.

XVII-5. THE CONSEQUENCES OF AN ATR ACCIDENT

The doses defined in 10 CFR-100 have been used as reference values. A dose of 25 rem integrated beta-gamma or 300 rem integrated iodine is used to evaluate the consequences of radioactive releases to the ATR environs.

1. CONSEQUENCES OF AN ATR ACCIDENT TO THE MTR-ETR-ATR AREA

There are three consequences to the MTR-ETR-ATR area from an accident at ATR. The first is the effect of an accident on personnel at the site before they are evacuated; the second is the time the site must remain evacuated until controlled short-time entry of MTR and ETR is possible; and the third is the length of time after an accident before relatively uninterrupted operation of MTR and ETR is possible.

The evacuation hazard has been determined in Sections XVII-4-4.2 and 4-3. These sections show that the maximum integrated doses received by persons outside the ATR building are approximately equal to or less than the reference doses of 25 rem of beta-gamma or 300 rem of iodine ingestion.

After the area has been evacuated, limited reentry to MTR and ETR will be possible when the direct gamma dose rate has fallen to an acceptable value. In general, the immersion beta-gamma and iodine ingestion doses will not be controlling because the time of entry can be limited to times of favorable weather conditions. The time until limited reentry will be permitted has been arbitrarily chosen as the time when the calculated dose rate for direct gamma is less than 5 rem per hour. The actual direct gamma dose will be considerably lower than the calculated one since no allowance has been made for shielding by the building structures. The use of the building sprays at ATR will shorten the time until safe reentry is permissible, because the removal of the iodine and solid fission products from the ATR building air will reduce the direct gamma dose rate as well as the air-borne activity. The results of the analysis are given in Table XVII-5-1.

TABLE XVII-5-1

ELAPSED TIME IN DAYS TO INITIAL REENTRY AFTER ATR ACCIDENTS
(based on 5 rem per hour direct gamma radiation)

Reactor	Fuel Transfer Accident		Complete Core Meltdown	
	Building Sprays Used	Sprays Not Used	Building Sprays Used	Sprays Not Used
MTR	< 1 hr	< 1 hr	3.0 days	7.5 days
ETR	<< 1 hr	<< 1 hr	0.5 days	4.0 days

The time periods shown in the table for the complete core meltdown exceed the greatest length of time that MTR and ETR have been evacuated to date. However, the reactors are designed to be safe for extended periods after shutdown and the potential hazard is probably reduced as the decay heat generation decreases with passing time.

Uninterrupted operation of MTR and ETR can be resumed when the beta-gamma and iodine dose rates decay to acceptable tolerances. These values have not been established, but the length of time to uninterrupted operation following the complete core meltdown can be inferred from Figures XVII-4-6 and -8. These show that under the worst weather conditions of Case A, with spray operation, the iodine dose rate falls to 0.1 rem per hour in 45 days, and in 70 days without sprays. Following the fuel transfer accident, the waiting time with Case A weather conditions is 25 days with sprays, and 46 days without sprays. These time periods are not appreciably greater than some shutdown periods for major maintenance at ETR in the past.

2. HAZARDS WITHIN NRTS BEYOND THE MTR-ETR-ATR AREA

The hazard to the working staff in the NRTS following an ATR accident is a function of their distance from ATR. In the event of an accident, the entire NRTS will be alerted in time to start appropriate emergency procedures. Following a major ATR accident, the decision to evacuate any area in the NRTS will depend on weather conditions and the severity of the accident.

After the complete core meltdown accident at ATR, the minimum time to accumulate a sufficient dose to require personnel evacuation is shown in Table XVII-5-2 for various distances. The times shown in this table are based on Cases A and A-S. The total time also included the effect of the ATR accident timetable, as stated in Section XVII-3 and the time for travel of the plume to the facility in question. Time for plume travel is based upon the inversion wind velocity listed in Table XVII-4-1 for Case A.

TABLE XVII-5-2

MINIMUM TIME TO EVACUATE FOLLOWING A COMPLETE CORE MELTDOWN AT ATR, CASES A AND A-S WEATHER

Distance	Time of Arrival of Plume in Hours After Time 0	Time in Hours to Receive 300 Rem of Iodine		Time in Hours to Receive 25 Rem of Beta Gamma		Total Time in Hours to Evacuation	
		Spray	No Spray	Spray	No Spray	Spray	No Spray
2 miles (ICFP)	1.02	0.52	0.33	0.68	0.64	1.5	1.35
5 miles (NRF, Spert, EBR-1, CFA, EBR-2)	2.11	∞	1.3	4.6	3.60	6.7	3.4
8 miles - Nearest NRTS Boundary	3.21	∞	3.0	26.0	12.0	29.2	6.2
10 miles (AREA)	3.94	∞	4.5	110	23.0	114	8.4
20 miles (ANP)	7.58	∞	30	∞	∞	∞	37.6
30 miles	11.2	∞	∞	∞	∞	∞	∞

The minimum evacuation time shown in Table XVII-5-2 will probably be greater under actual weather conditions. The factors which increase the time are as follows:

- (1) Inversion conditions, where $n = 0.5$ occur only 15 percent of the time
- (2) The maximum inversion wind direction frequency is 22 percent
- (3) Facilities may not be located on the axis of the plume.

After a fuel transfer accident, the times shown in Table XVII-5-2 will be approximately 33-fold greater because of the reduced activity release.

The information presented in Table XVII-5-2 shows that there is ample time to evacuate personnel at any of the facilities on the axis of the plume for either accident. At the more distant sites it is unlikely that personnel would receive a total dose of 300 rem even if they are not evacuated.

3. AREA ACCESSIBILITY AFTER ACCIDENT BEYOND MTR-ETR AREA

These operating areas are beyond the influence of the direct radiation from the ATR building, but during unfavorable weather conditions may be within the plume of gas from the building.

For the purposes of this analysis, it was assumed that uninterrupted operation can be resumed in a facility if the calculated dose rate from iodine is less than 0.1 rem per hour or if the dose rate from beta-gamma immersion is less than 0.01 rem per hour. The time period to reach these dose rates is shown in Table XVII-5-3 for Case A weather conditions, with and without sprays.

TABLE XVII-5-3

TIME TO RESUME CONTINUOUS OPERATION FOLLOWING AN ACCIDENT AT ATR -- DAYS

Distance	Fuel Transfer Accident		Complete Core Meltdown	
	Sprays Used	Sprays Not Used	Sprays Used	Sprays Not Used
2 miles (ICPP)	5.6	28	21	46
5 miles (NRF, Spert, EBR-I, CFA, EBR-II)	1.5	20	15	40
10 miles (AREA)	0.7	15	12	35

There is no hazard involved in interrupting operations from time to time at the various NRTS sites because each operation has established procedures

for safe shutdown. However, frequent or extended shutdowns can seriously delay programs under way at the various sites.

4. HAZARDS TO AREAS OUTSIDE NRTS

4.1 Low Population Area

The existence of unfavorable (Case A) weather conditions at the time of the accident may make it necessary to evacuate some of the closer communities. The minimum time for evacuation of these closer communities with Case A weather can be obtained from Table XVII-5-2. Even without sprays, the available time before evacuation must be completed is greater than six hours for the point on the NRTS boundary nearest ATR. This leaves ample time to organize and carry out post-emergency evacuations. The probability is high that variation in wind and weather will be such during the time following a complete core meltdown that the evacuation of off-site persons will not be necessary. The decision to evacuate will be based on the measurement of plume activity and the weather conditions at the time of the accident.

Case A weather is not representative of the actual conditions which can exist over an extended period of time. However, Cases II and II-S represent a reasonably conservative selection of weather conditions which neglects variation in wind direction and the effects of terrain irregularities on diffusion, and furthermore, takes no credit for radioactive decay in the plume en route, nor for fallout or rainout. Based on Case II weather, no point beyond 31 miles from ATR will receive an integrated dose of more than 25 rem of beta-gamma or 300 rem of iodine. The use of sprays (Case II-S) reduces the distance to eight miles. Table II-1-1 and Figure II-1-1 show a total urban population of 2391 living in the 31-mile zone and none in the 8-mile zone.

5. CONCLUSION

On the basis of the foregoing analysis, using the extreme assumption of a complete core meltdown, the ATR does not result in an unacceptable hazard to the environs because:

- (1) The MTR-ETR-ATR area can be evacuated without any personnel receiving an excessive integrated dose of either total beta-gamma or iodine except possibly for personnel in the ATR building.
- (2) Sufficient time is available to evacuate other areas on the NRTS and any of the small neighboring communities which may require evacuation.
- (3) Reentry to evacuated facilities other than the ATR can be achieved in a reasonable period of time.

(4) The potential dose at the nearest large population center is less than the reference values of 25 rem beta-gamma whole body dose and 300 rem iodine ingestion dose using extremely conservative assumptions in the calculations.

(5) The use of building sprays significantly reduces the above doses.

XVII-6. ENVIRONMENTAL HAZARDS ASSOCIATED WITH LARGE RELEASES OF RADIOACTIVITY IN THE NON-GASTIGHT AREA

Fuel handling accidents in the ATR canal system outside the gastight area have been examined.

Removal of a spent fuel element out of the canal by an operator error is considered incredible for the following reasons:

- (1) Fuel handling in the canal is performed manually by operators using long handled tools. The raising of an element out of the water onto the floor adjacent to the canal would require extreme if not impossible effort because of the fuel element weight and moment involved.
- (2) If an element is raised by error to the canal surface, radiation alarms would alert an operator of the hazard.
- (3) Training and administrative procedures ensure that all personnel working around the canal are aware of the potential hazards.

If an ATR fuel element is dropped from the vertical position to a flat position during handling operations in the storage canal, the normal natural convection heat transfer mechanism is hindered. Boiling and possibly melting of inside fuel plates could occur.

Methods to analyze this mode of heat transfer are uncertain especially if two-phase conditions are included. Based on MTR experience in which fuel elements are transferred from the reactor to the canal in the horizontal position during the latter phase of the operation, a qualitative comparison can be made. MTR elements have been discharged from the reactor in less than three hours following shutdown and held in a horizontal position for periods of more than one hour in some cases without observable nucleation. The maximum ATR operating heat flux is 2.23×10^6 Btu/hr-ft², compared to 8×10^5 Btu/hr-ft² in the MTR. The ATR fuel elements will not be discharged for 24 hours following reactor shutdown. Based on measured decay heat values for MTR elements [11], the ratio of decay heat values between 2.4 and 24 hours is 1.7. The decay heat flux for ATR fuel elements removed 24 hours after shutdown is, therefore, approximately 1.6 times the heat flux for MTR elements removed 2.4 hours after shutdown. The ATR fuel elements are also vented along the side plates providing additional flow paths for natural circulation.

In summary, MTR fuel elements lying flat in the canal do not exhibit observable nucleation. ATR vented fuel elements generate approximately 1.6 times the decay heat of the MTR elements and are not expected to melt.

The probability of dropping an ATR element during handling is small because of positive latching tools.

If it is assumed that melting occurs, only a fraction of the element is involved. Spent Destructive Test data [12, 13] show that less than 20 percent of the noble gases and less than 3 percent of the halogens would be released from the canal water.

The fission products released from the canal water would be discharged to the atmosphere via the building roof ventilators.

It is concluded that the released fission product inventory coupled with building downwash diffusion of the released cloud results in lower integrated doses to the TRA evacuation area than the fuel element melting accident analyzed as occurring in the gastight area.

XVII-7. REFERENCES

1. D. R. deBoisblanc et al, Hazards Evaluation of Proposed Site for ETR II, Phillips Petroleum Company, IDO-16623 (August 1960).
2. "Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure", U. S. Department of Commerce, National Bureau of Standards, Handbook 69, June 5, 1959.
3. Reactor Site Criteria, Notice of Proposed Guides, Federal Register, U. S. Atomic Energy Commission, 10 CFR Part 100, May 23, 1959.
4. Meteorology and Atomic Energy, U. S. Department of Commerce, Weather Bureau for U. S. Atomic Energy Commission, AECU-3066, July 1955.
5. G. A. DeMarrais and N. F. Islitzer, Diffusion Climatology of the National Reactor Testing Station, United States Weather Bureau, Idaho Falls, Idaho, IDO-12015 (April 1960).
6. Nuclear Incident at the Idaho Chemical Processing Plant, Phillips Petroleum Company, IDO-10036 (January 25, 1961).
7. V. Griffiths, The Removal of Iodine From the Atmosphere by Sprays, AHSB(S)R-45 (December 1962).
8. L. C. Watson et al, Iodine Containment by Dousing in NPD-II, CRCE-979 (October 1960).
9. J. J. DiNunno et al, Calculation of Distance Factors for Power and Test Reactor Sites, TID-14844 (1962).
10. Weather Bureau, Washington, D. C., Meteorology and Atomic Energy, AECU-3066 (July 1955).
11. W. C. Francis and L. L. Marsden, Experimental and Theoretical Values of the Gamma Decay Dose Rate and Heating from Spent MTR Fuel Elements, IDO-16247 (January 1956).
12. D. Bunch, Spert I Destructive Test Series Environmental and Research Studies, IDO-12039 (January 1965).
13. D. Bunch, "Measurements of Released Fission Products from a Reactor Excursion". Paper presented at International Symposium on Fission-Product Release and Transport Under Accident Conditions -- Oak Ridge, Tennessee, April 5-7, 1965.

APPENDIX A

IRRADIATION EFFECTS ON BERYLLIUM

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX A

IRRADIATION EFFECTS ON BERYLLIUM

The beryllium reflector blocks in the ATR have a design life requirement of five years operation. Existing experimental data on the performance of beryllium in test reactor applications have indicated possible failure. The MTR beryllium has the highest integrated dose achieved to date and equivalent to exposure of the ATR beryllium for one year. A discussion is presented below on factors that could possibly affect the properties of the beryllium reflector for the higher burnups to be experienced in the ATR.

The structural properties of beryllium are altered by irradiation principally by two phenomena: (a) atomic displacements associated with neutron-atom collisions and (b) gases induced by beryllium transmutations.

Atom displacements are primarily responsible for sharp increases in tensile and yield strength and corresponding reductions in ductility and toughness of irradiated material. The rate of property change decreases with exposure time, and the effects of irradiation are said to "saturate". "Saturation" in beryllium irradiated at 100°F occurs at about 1×10^{21} n/cm² (Reference 1) greater than 1 MeV, equivalent to about 40 days of ATR operation. The beryllium ductility before irradiation is approximately 1.5 percent total tensile elongation, and is nil after the above exposure. Thus, the highly irradiated material will not accommodate plastic strain in tension. The properties in compression after irradiation are not known but are expected to be somewhat better than the tensile properties. Therefore, stress concentrations and high tensile stresses have been avoided in the design of the ATR beryllium components.

The second effect of irradiation is the accumulation of helium and tritium produced by beryllium transmutations. Although several transmutations are possible, the (n, 2n) reaction is the most important, since this reaction is responsible for about 90 percent of the total gas yield. In this reaction, two helium atoms are formed for each beryllium atom destroyed. Since the gases produced must be accommodated by the beryllium lattice, this phenomenon is responsible for further changes in the properties of the beryllium. Unlike the atomic displacement damage, these reactions continue at a rate proportional to the fast flux and do not "saturate". From the standpoint of permanent damage to the beryllium, the gas could produce the following effects:

- (1) Cracking from internal stresses produced by the accumulation of the additional atoms in the lattice.
- (2) Loss of beryllium crystal structure which produces granulation.
- (3) Gross swelling resulting from the diffusion and agglomeration of the gases at imperfections in the crystals and grain boundaries.
- (4) Growth resulting from the accumulation of helium atoms in the beryllium lattice.

Items 1 and 2 are purely speculative since there is no experimental evidence that such phenomena will occur. It is believed that an exposure greater than the five year nuclear lifetime of the ATR beryllium would be required before cracking or granulation would become significant. For example, after five years ATR operation, only about 1.5 percent of the beryllium atoms would be destroyed in the peak flux areas. The helium atoms produced would have an average spacing of about 8 Å apart. According to Mott and Nabarro [2], the strain at a distance R from a helium atom would be proportional to $(r_0/R)^3$ where r_0 is the radius of the helium atom (0.83 Å). Thus, midway between two adjacent helium atoms, the strain due to each atom would be on the order of 1/100 of the strain immediately surrounding the helium atom. Reinforcement of strain due to overlapping of two adjacent stress fields would appear to be insignificant. It is concluded that after five years exposure the maximum internal strain due to the induced helium atoms would not be greater than the strain associated with a single isolated atom. Therefore, spallation due to internal strains should not occur even though the average strain in the beryllium may increase due to slight overlapping of individual atom effects. Since only one out of every 15 unit cells contains a helium atom after five years' exposure, the beryllium structure will be retained and no granulation will result.

With respect to item 3, data [3] indicate that beryllium must be irradiated at temperatures considerably higher than the ATR temperature (250°F) before gas diffusion, agglomeration, and swelling occur. In the case of an irradiation exposure equivalent to the production of 25 cc of gas per cc of beryllium, post annealing experiments reveal that temperatures above 1200°F are required to produce swelling. Temperatures above 1400°F are required for swelling specimens containing 5 cc of gas per cc of beryllium. Assuming a linear extrapolation, the temperature for onset of swelling of the ATR beryllium after five years' operation (90 cc gas per cc beryllium) would be greater than 500°F. Therefore, it appears improbable that irradiation induced swelling would occur in the ATR beryllium during the five-year operation.

The primary consideration associated with item 4 is the effect of growth on the dimensions of the reflector components and distortion of the components due to varying growth rates through the material. Since experimental data on the irradiation-induced growth of beryllium are not available, the growth of beryllium as a function of neutron flux and time was estimated on this basis.

The growth at any given time may be expressed as the product of two factors: a growth factor and a reaction factor. Since the $(n, 2n)$ transmutation accounts for about 90 percent of the total gas yield, the reaction factor was computed for this reaction only. The reaction factor employed predicts gas yields within 10 percent of experimental measurements.

The two helium atoms resulting from the $(n, 2n)$ reaction must be accommodated by their beryllium lattice. Some of the helium atoms will occupy interstitial locations and others will occupy lattice sites. The split is not determinable but it is believed that most of the gas atoms will occupy interstitial locations. However, the substitutional condition should produce the largest volume change since additional lattice sites would be generated. Thus, in order to be on the conservative side, it was assumed that all the helium atoms form a substitutional solid solution by forming additional lattice sites. It was also assumed that the destruction of a beryllium atom leaves a vacancy, and the difference in size of

the helium and beryllium atoms was reflected in the computation. On this basis, the linear growth factor was computed to be 0.22 linear percent growth per atom percent reaction.

The resulting expression was used to plot growth as a function of neutron flux in Figure A-1. Unfortunately, experimental data which could be used for correlation are not available. Since the growth of the MTR beryllium predicted by this expression is within the accuracy of hot cell or pool measurements, growth of this magnitude would not be detected. This agrees with findings by Phillips Petroleum Company and also agrees with density data [1] which show that volume changes on MTR beryllium pieces have not been detected.

FIG. A-1 IRRADIATION-INDUCED GROWTH OF BERYLLIUM.

It is recognized that the foregoing discussion is not a substitute for experimental data. Accordingly, beryllium surveillance specimens will be placed in

regions of the core where their exposures would lead the reflector components by a comfortable margin as determined by the frequency of specimen withdrawal (ie, the longer the time between withdrawals the greater the lead accumulated). It is planned to preexpose the surveillance specimens in the MTR or ETR to accumulate a margin of n/cm^2 -- prior to insertion in the ATR. In this way, it should be possible to monitor in advance the pertinent properties of the ATR reflector for nearly five years of reactor operation. The properties of interest are low-temperature growth, density, tensile properties, and resistance to cracking due to internal stressing.

In summary, the present state of the technology assures a beryllium life of more than one year and a possible life of five years or more depending on the results of periodic inspections of the reflector and the experimental data from the surveillance specimens.

REFERENCES

1. M. H. Bartz, Performance of Metals During Six Years Service in the MTR, Geneva Conference Paper, 1958.
2. N. F. Mott and F. R. N. Nabarro, "Report of a Conference on Internal Strains in Solids", The Proceedings of the Physical Society, 52, Part 1, January 1940.
3. C. E. Ells and E. C. W. Perryman, "Effects of Neutron-Induced Gas Formation on Beryllium", Journal of Nuclear Materials, 1, No. 1, (1959).

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX B

ENVIRONMENTAL HAZARDS

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX B

ENVIRONMENTAL HAZARDS

1. NOMENCLATURE FOR ENVIRONMENTAL HAZARDS ANALYSIS

<u>Symbols</u>	<u>Definitions</u>
D	= Integrated dose in rem
$\left(\frac{D_I}{X}\right)_i$	= Iodine dose rate per unit concentration in $\frac{\text{rem/sec}}{\text{curie/cu m}}$
E_β	= Average beta energy for an isotope in MeV/dis
E_γ	= Average gamma energy for an isotope in MeV/dis
F_p	= Fraction of isotope released from primary system
F_b	= Fraction of released isotope which remains airborne
Q	= Total number of curies of an isotope released from the building during the accident
q	= Initial number of curies of an isotope in the core at the time of the accident
R	= Dose rate in rem/hour
S	= Factor by which building sprays reduce the integrated dose
s	= Spray factor; the fraction of an isotope not removed by the building sprays
t	= Time interval since the start of the release in sec^{-1}
t_s	= The time period during which the building sprays are operating in sec
V_b	= Volume of gastight portion of building in ft^3
V_r	= Volume of reactor room in ft^3
X	= Concentration of an isotope in curies/cu m
$\left(\frac{X}{Q}\right)$	= Atmospheric dilution factor based on Sutton's equation in $\frac{\text{curies/cu m}}{\text{curies/sec}}$
λ_b	= Building leakage constant corresponding to 10 percent of the building volume per day, in sec^{-1}
λ_i	= Effective radioactive decay constant, sec
ρ_o	= Density of atmosphere at sea level

SubscriptsDefinitions

ρ = Density of atmosphere at the site

$\sum_{i=1}^i$ = Summation of an i number of doses or dose rates corresponding to each isotope

b Refers to the building (gastight volume)

I Refers to iodine isotopes only

i Refers to each individual isotope

m Refers only to the mixed fission products

p Refers to the primary system

β Refers to beta energy, dose, or dose rate

γ Refers to gamma energy, dose, or dose rate

Super-
scripts

'(prime) Refers to the use of building sprays

2. GENERAL DATA

$$\lambda_b = \frac{0.1 \text{ (days}^{-1}\text{)} V}{86,400 \frac{\text{sec}}{\text{day}} V} = 1.16 \times 10^{-6} \text{ sec}^{-1}$$

$$V_s = 517,000 \text{ ft}^3$$

$$V_\beta = 1,640,000 \text{ ft}^3$$

Isotope	Effective Half-Life	Effective Decay Constant (λ_1)	$(\lambda_1 + \lambda_b)$	$\left(\frac{\lambda_b}{\lambda_1 + \lambda_b}\right)$	q_1
I-131	8.08 d	0.993×10^{-6}	2.15×10^{-6}	0.540	4.68×10^6
I-132 (Te-132)	77.7 h	2.48×10^{-6}	3.64×10^{-6}	0.318	9.10×10^6
I-133 (Te-132)	21 h	0.916×10^{-5}	1.03×10^{-5}	0.113	1.37×10^7
I-134	53 m	2.18×10^{-4}	2.19×10^{-4}	0.005	1.61×10^7
I-135	6.7 h	2.88×10^{-5}	3.00×10^{-5}	0.039	1.25×10^7
Xe-131m	12 d	0.666×10^{-6}	1.82×10^{-6}	0.64	1.87×10^5
Xe-133m	2.3 d	0.347×10^{-5}	0.463×10^{-5}	0.25	3.27×10^5
Xe-133	5.27 d	0.152×10^{-5}	0.268×10^{-5}	0.43	1.20×10^7
Xe-135m	15.6 m	0.738×10^{-3}	0.739×10^{-3}	0.0015	3.74×10^6
Xe-135	9.13 h	0.210×10^{-4}	0.221×10^{-4}	0.050	1.38×10^7
Kr-83m	114 m	1.02×10^{-4}	1.03×10^{-4}	0.011	1.02×10^6
Kr-85m	4.36 h	0.440×10^{-4}	0.45×10^{-4}	0.026	3.17×10^6
Kr-87	78 m	1.47×10^{-4}	1.48×10^{-4}	0.008	5.70×10^6
Kr-88	2.77 h	0.693×10^{-4}	0.704×10^{-4}	0.016	7.82×10^6
Mixed F. P.	---	---	---	0.211	8.84×10^8

Isotope	F_b	F_p	$\left(\frac{D_I}{X}\right)_i$	E_β	E_γ	s_i	S_i
I-131	0.5	0.5	329	0.44	0.4	0.01	0.084
I-132 (Te-132)	0.5	0.5	12.4	1.7	0.8	0.01	0.132
I-133	0.5	0.5	92.3	0.84	0.55	0.01	0.317
I-134	0.5	0.5	5.66	0.82	1.3	0.01	1.000
I-135	0.5	0.5	25.3	1.2	1.5	0.01	0.665
Xe-131m	1.0	1.0	---	0.44	0.163	1.00	1.00
Xe-133m	1.0	1.0	---	0.84	0.233	1.00	1.00
Xe-133	1.0	1.0	---	0.17	0.081	1.00	1.00
Xe-135m	1.0	1.0	---	1.2	0.520	1.00	1.00
Xe-135	1.0	1.0	---	0.5	0.250	1.00	1.00
Kr-83m	1.0	1.0	---	0.01	0.02	1.00	1.00
Kr-85m	1.0	1.0	---	0.44	0.20	1.00	1.00
Kr-87	1.0	1.0	---	2.8	2.00	1.00	1.00
Kr-88	1.0	1.0	---	0.2	2.00	1.00	1.00
Mixed F. P.	1.0	0.01	---	0.4	0.70	0.25	0.38

3. INTEGRATED IODINE DOSE TO THE THYROID

$$D_I = \sum_1^i Q_i \left(\frac{D_I}{X} \right)_i \left(\frac{X}{Q} \right)$$

where

$$Q_i = \lambda_b F_b F_p q_i \int_0^{\infty} e^{-(\lambda_i + \lambda_b)t} dt = F_b F_p q_i \left[\frac{\lambda_b}{\lambda_i + \lambda_b} \right]$$

$$D_I = \left(\frac{X}{Q} \right) \sum_1^i F_b F_p q_i \left[\frac{\lambda_b}{\lambda_i + \lambda_b} \right] \left(\frac{D_I}{X} \right)_i$$

If building sprays are used, the iodine dose will be reduced by the factor S_i .

$$D'_I = \sum_1^i S_i Q_i \left(\frac{D_I}{X} \right)_i \left(\frac{X}{Q} \right)$$

where

$$S_i = (1 - s_i) \left[1 - e^{-(\lambda_i + \lambda_b)t_s} \right] + s_i$$

4. IODINE DOSE RATE TO THE THYROID

$$R_I = 3600 \frac{\text{sec}}{\text{hr}} \sum_1^i \left(\frac{dQ}{dt} \right)_i \left(\frac{D_I}{X} \right)_i \left(\frac{X}{Q} \right)$$

where

$$\left(\frac{dQ}{dt} \right)_i = \lambda_b F_b F_p q_i e^{-(\lambda_i + \lambda_b)t}$$

$$R_I = 3600 \frac{\text{sec}}{\text{hr}} \left(\frac{X}{Q} \right) \sum_1^i \lambda_b F_b F_p q_i \left(\frac{D_I}{X} \right)_i e^{-(\lambda_i + \lambda_b)t}$$

If building sprays are used, the iodine dose rate will be reduced by the spray factor (s).

$$R_I' = s R_I$$

where

$$s = e^{-1.28 \times 10^{-4} t} \quad 0 < t < 36,000 \text{ sec.}$$

$$s = 0.01 \quad 36,000 < t < \infty \text{ sec.}$$

5. INTEGRATED BETA DOSE FROM THE PLUME

$$D_\beta^{[a]} = \sum_1^i \frac{0.5 \times 0.64 \times 3.7 \times 10^{10}}{6.8 \times 10^{10}} \frac{\rho_o}{\rho} E_\beta \int_0^\infty X_\beta dt$$

$$D_\beta = \sum_1^i \frac{0.5 \times 0.64 \times 3.7 \times 10^{10} \left(\frac{\text{dis/sec}}{\text{curie}} \right)}{6.8 \times 10^{10} \left(\frac{\text{MeV/m}^3}{\text{rem}} \right)} \frac{\rho_o}{\rho} E_\beta \left(\frac{X}{Q} \right) Q_i$$

where

$$Q_i = \lambda_b F_b F_p q_i \int_0^\infty e^{-(\lambda_i + \lambda_b)t} dt = F_b F_p q_i \left[\frac{\lambda_b}{\lambda_i + \lambda_b} \right]$$

$$D_\beta = 0.212 \left(\frac{X}{Q} \right) \sum_1^i E_\beta F_b F_p q_i \left[\frac{\lambda_b}{\lambda_i + \lambda_b} \right]$$

If building sprays are used, the beta dose will be reduced by the factor (S).

$$D_\beta' = 0.212 \left(\frac{X}{Q} \right) \sum_1^i S_i E_\beta F_b F_p q_i \left[\frac{\lambda_b}{\lambda_i + \lambda_b} \right]$$

where

$$S_i = (1 - s_i) \left[1 - e^{-(\lambda_i + \lambda_b)ts} \right] + s_i$$

[a] Reference 1

Since the mixed fission products do not have a decay constant, the above equations for Q_i and S_i are not applicable for the mixed fission products. However, the decay function for the mixed fission products was graphically integrated to obtain approximate solutions for Q_m and S_m .

$$Q_m = F_b F_p q_m [0.211]$$

$$S_m = (1 - 0.25) [0.173] + 0.25 = 0.38 \text{ for } t_s = 36,000 \text{ sec.}$$

6. BETA DOSE RATE FROM THE PLUME

$$R_\beta = \frac{dD_\beta}{dt} = \sum_i 3600 \frac{\text{sec}}{\text{hr}} \frac{0.5 \times 0.64 \times 3.7 \times 10^{10} \left(\frac{\text{dis/sec}}{\text{curie}} \right)}{6.8 \times 10^{10} \frac{\text{MeV/m}^3}{\text{rem}}} \frac{\rho_o}{\rho} E_\beta \left(\frac{dQ}{dt} \right)_i \left(\frac{X}{Q} \right)$$

$$R_\beta = \sum_i 763 E_\beta \left(\frac{dQ}{dt} \right)_i \left(\frac{X}{Q} \right)$$

where

$$\frac{dQ}{dt} = \lambda_b F_b F_p q_i e^{-(\lambda_i + \lambda_b)t}$$

$$R_\beta = 763 \times \left(\frac{X}{Q} \right) \sum_i \lambda_b F_b F_p q_i E_\beta e^{-(\lambda_i + \lambda_b)t}$$

If building sprays are used, the beta dose rate will be reduced by the spray factor (s).

$$R_\beta = \sum_i s_i R_\beta = 763 \times \left(\frac{X}{Q} \right) \sum_i s_i \lambda_b F_b F_p q_i E_\beta e^{-(\lambda_i + \lambda_b)t}$$

where

$$0 < t < 36,000 \text{ sec}$$

$$s (\text{iodine}) = e^{-1.28 \times 10^{-4} t}$$

$$s (\text{noble gases}) = 1.00$$

$$s (\text{mixed fission products}) = e^{-3.85 \times 10^{-5} t}$$

$$36,000 < t < \infty \text{ sec}$$

$$s (\text{iodine}) = 0.01$$

$$s (\text{noble gases}) = 1.00$$

$$s (\text{mixed fission products}) = 0.25.$$

The value of $e^{-(\lambda_i + \lambda_b)t}$ for the mixed fission products was obtained by plotting the normalized decay curve for the mixture (the effective $e^{-\lambda t}$) and multiplying it by the factor $e^{-\lambda_b t}$.

7. INTEGRATED GAMMA DOSE FROM THE PLUME

$$D_Y^{[a]} = \sum_{i=1}^i \frac{3.7 \times 10^{10} \frac{\text{dis/sec}}{\text{curie}}}{6.8 \times 10^{10} \frac{\text{MeV/m}^3}{\text{rem}}} E_Y \frac{\text{MeV}}{\text{dis}} Q_i \left(\frac{X}{Q} \right)$$

$$Q_i = \lambda_b F_b F_p q_i \int_0^{\infty} e^{-(\lambda_i + \lambda_b)t} dt = F_b F_p q_i \left[\frac{\lambda_b}{\lambda_i + \lambda_b} \right]$$

$$D_Y = 0.547 \left(\frac{X}{Q} \right) \sum_{i=1}^i E_Y Q_i = 0.547 \left(\frac{X}{Q} \right) \sum_{i=1}^i E_Y F_b F_p q_i \left[\frac{\lambda_b}{\lambda_i + \lambda_b} \right]$$

If building sprays are used, the gamma dose will be reduced by the factor S_i .

$$D_Y' = 0.547 \left(\frac{X}{Q} \right) \sum_{i=1}^i S_i E_Y F_b F_p q_i \left[\frac{\lambda_b}{\lambda_i + \lambda_b} \right]$$

where

$$S_i = (1 - s_i) \left[1 - e^{-(\lambda_i + \lambda_b)t_s} \right] + s_i$$

Since the mixed fission products do not have a decay constant, the above equations for Q_i and S_i are not applicable for the mixed fission products. However, the decay function for the mixed fission products was graphically integrated to obtain approximate solutions for Q_m and S_m .

$$Q_m = F_b F_p q_m [0.211]$$

$$S_m = (1 - 0.25) [0.173] + 0.25 = 0.38 \quad \text{for } t_s = 36,000 \text{ sec.}$$

[a] Reference 1.

8. GAMMA DOSE RATE FROM THE PLUME

$$R_Y = \frac{dD_Y}{dt} = \sum_i 3600 \frac{\text{sec}}{\text{hr}} \frac{3.7 \times 10^{10} \frac{\text{dis/sec}}{\text{curie}}}{6.8 \times 10^{10} \frac{\text{MeV/m}^3}{\text{rem}}} E_Y \frac{\text{MeV}}{\text{dis}} \left(\frac{dQ}{dt} \right)_i \left(\frac{X}{Q} \right)$$

$$R_Y = \sum_i 1970 E_Y \left(\frac{dQ}{dt} \right)_i \left(\frac{X}{Q} \right)$$

where

$$\frac{dQ}{dt} = \lambda_b F_b F_p q_i e^{-(\lambda_i + \lambda_b)t}$$

$$R_Y = 1970 \times \left(\frac{X}{Q} \right) \sum_i E_Y \lambda_b F_b F_p q_i e^{-(\lambda_i + \lambda_b)t}$$

If building sprays are used, the gamma dose rate will be reduced by the spray factor (s).

$$R'_Y = \sum_i s_i R_Y = 1970 \times \left(\frac{X}{Q} \right) \sum_i s_i E_Y \lambda_b F_b F_p q_i e^{-(\lambda_i + \lambda_b)t}$$

where

$0 < t < 36,000 \text{ sec.}$

$$\begin{aligned} s \text{ (iodine)} &= e^{-1.28 \times 10^{-4} t} \\ s \text{ (noble gases)} &= 1.00 \\ s \text{ (mixed fission products)} &= e^{-3.85 \times 10^{-5} t} \end{aligned}$$

$36,000 < t < \infty \text{ sec.}$

$$\begin{aligned} s \text{ (iodine)} &= 0.01 \\ s \text{ (noble gases)} &= 1.00 \\ s \text{ (mixed fission products)} &= 0.25 \end{aligned}$$

The value of $e^{-(\lambda_i + \lambda_b)t}$ for the mixed fission products was obtained by plotting the normalized decay curve for the mixture (the effective $e^{-\lambda_i t}$) and multiplying it by the factor $e^{-\lambda_b t}$.

9. DIRECT GAMMA DOSE RATE

The direct gamma dose rates resulting from a complete core meltdown were calculated using an existing computer code. The reduced dose rates corresponding to the use of building sprays were also calculated.

The program did not correct the calculated dose rates for the gamma-ray attenuation due to the building structures, the leakage of activity from the building, nor the distribution of the source throughout the gastight volume of the building.

In order to correct the dose rates for the reduction of the source strength due to building leakage and for the fraction of the source above grade, the code results were multiplied by a correction factor, f .

$$f = \frac{V_r}{V_b} e^{-\lambda_1 t}$$

10. REFERENCE

1. Meteorology and Atomic Energy, U. S. Department of Commerce, Weather Bureau for U. S. Atomic Energy Commission, AECU-3066, July 1955.

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX C

ATR FISSION PRODUCT INVENTORY AFTER 17 DAYS OF OPERATION

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX C

ATR FISSION PRODUCT INVENTORY AFTER 17 DAYS OF OPERATION

TABLE C-I

ATR FISSION PRODUCT INVENTORY AFTER 17 DAYS OPERATION

250 MW		33 kg of Fuel					
Note: Numbers in parentheses are exponents of ten.							
Disintegrations per Second per Gram of Uranium							
Isotope	Time After Shutdown						
	0	1/2 hour	1 hour	12 hours	24 hours	3 days	90 days
Se-83m	7.10 (11)	5.86 (3)					
Se-83	4.97 (11)	2.17 (11)	9.48 (10)	1.07 (3)			
Br-83	1.14 (12)	1.03 (12)	9.04 (11)	3.36 (10)	9.05 (8)	4.73 (2)	
Kr-83m	1.14 (12)	1.13 (12)	1.10 (12)	1.17 (11)	4.24 (9)	2.70 (3)	
Se-84	2.60 (12)	7.96 (7)	2.43 (3)				
Br-84	2.60 (12)	1.39 (12)	6.97 (11)	1.66 (5)			
Br-85	3.55 (12)	3.47 (9)					
Kr-85m	3.55 (12)	3.31 (12)	3.06 (12)	5.33 (11)	7.92 (10)	3.84 (7)	
Kr-85	2.14 (9)	2.14 (9)	2.14 (9)	2.16 (9)	2.17 (9)	2.17 (9)	2.13 (9)
Br-87	6.39 (12)	9.03 (2)					
Kr-87	6.39 (12)	4.95 (12)	3.79 (12)	1.08 (10)	1.80 (7)		
Br-88	6.87 (12)						
Kr-88	8.76 (12)	7.73 (12)	6.82 (12)	4.35 (11)	2.16 (10)	1.32 (5)	
Rb-88	8.76 (12)	8.33 (12)	7.54 (12)	4.88 (11)	2.42 (10)	1.47 (5)	
Br-89	1.09 (13)						
Kr-89	1.09 (13)	1.61 (10)	2.33 (7)				
Rb-89	1.14 (13)	3.69 (12)	9.58 (11)				
Sr-89	2.26 (12)	2.26 (12)	2.26 (12)	2.25 (12)	2.23 (12)	2.17 (12)	6.98 (11)
Kr-90	1.23 (13)						
Rb-90	1.40 (13)	8.64 (9)	4.38 (6)				
Sr-90	1.62 (10)	1.61 (10)	1.61 (10)	1.61 (10)	1.61 (10)	1.61 (10)	1.60 (10)
Y-90	1.25 (10)	1.25 (10)	1.25 (10)	1.29 (10)	1.33 (10)	1.44 (10)	1.60 (10)
Kr-91	8.76 (12)						
Rb-91	1.35 (13)	3.07 (12)	6.97 (11)				
Sr-91	1.40 (13)	1.37 (13)	1.32 (13)	6.06 (12)	2.57 (12)	8.33 (10)	
Y-91m	5.59 (12)	5.57 (12)	5.51 (12)	2.66 (12)	1.12 (12)	3.65 (10)	
Y-91	2.48 (12)	2.47 (12)	2.47 (12)	2.50 (12)	2.52 (12)	2.49 (12)	8.81 (11)
Kr-92	6.39 (12)						
Rb-92	1.30 (13)	2.24 (6)					
Sr-92	1.44 (13)	1.27 (13)	1.11 (13)	5.94 (11)	2.42 (10)	6.73 (4)	
Y-92	1.44 (13)	1.43 (13)	1.41 (13)	3.64 (12)	4.52 (11)	4.98 (7)	

TABLE C-I (cont.)

ATR FISSION PRODUCT INVENTORY AFTER 17 DAYS OPERATION

Isotope	Disintegrations per Second per Gram of Uranium						
	Time After Shutdown						
	0	1/2 hour	1 hour	12 hours	24 hours	3 days	90 days
Kr-93	3.08 (12)						
Rb-93	1.04 (13)						
Sr-93	1.52 (13)	7.78 (11)	3.99 (10)				
Y-93	1.54 (13)	1.50 (13)	1.45 (13)	6.77 (12)	2.95 (12)	1.06 (11)	
Kr-94	1.42 (12)						
Rb-94	6.87 (12)						
Sr-94	1.37 (13)	4.21 (8)	1.28 (4)				
Y-94	1.54 (13)	4.90 (12)	1.39 (12)	1.27 (0)			
Y-95	1.51 (13)	2.09 (12)	2.88 (11)				
Zr-95	2.51 (12)	2.51 (12)	2.51 (12)	2.49 (12)	2.48 (12)	2.43 (12)	9.62 (11)
Nb-95m	2.49 (10)	2.49 (10)	2.49 (10)	2.57 (10)	2.65 (10)	2.89 (10)	1.42 (10)
Nb-95	3.87 (11)	3.87 (11)	3.88 (11)	4.07 (11)	4.27 (11)	5.06 (11)	1.22 (12)
Zr-97	1.47 (13)	1.43 (13)	1.40 (13)	9.00 (12)	5.51 (12)	7.80 (11)	
Nb-97m	1.40 (13)	1.36 (13)	1.34 (13)	8.55 (12)	5.24 (12)	7.42 (11)	
Nb-97	1.46 (13)	1.46 (13)	1.45 (13)	9.68 (12)	5.94 (12)	8.39 (11)	
Nb-99	1.44 (13)	6.08 (10)	2.56 (8)				
Mo-99	1.42 (13)	1.41 (13)	1.40 (13)	1.25 (13)	1.11 (13)	6.83 (12)	3.94 (3)
Tc-99m	1.42 (12)	1.41 (12)	1.41 (12)	1.34 (12)	1.21 (12)	7.49 (11)	4.32 (2)
Mo-101	1.18 (13)	2.85 (12)	6.86 (11)				
Tc-101	1.18 (13)	6.78 (12)	2.52 (12)				
Mo-102	9.94 (12)	1.75 (12)	3.10 (11)				
Tc-102	9.94 (12)	1.82 (12)	3.22 (11)				
Ru-103	1.76 (12)	1.75 (12)	1.75 (12)	1.74 (12)	1.72 (12)	1.67 (12)	3.67 (11)
Rh-103m	1.67 (12)	1.67 (12)	1.67 (12)	1.65 (12)	1.64 (12)	1.58 (12)	3.49 (11)
Mo-105	1.42 (12)	2.22 (10)	3.47 (8)				
Tc-105	2.13 (12)	2.22 (10)	3.84 (8)				
Ru-105	2.13 (12)	1.99 (12)	1.84 (12)	3.39 (11)	5.35 (10)	3.29 (7)	
Rh-105m	2.13 (12)	2.00 (12)	1.85 (12)	3.40 (11)	5.36 (10)	3.30 (7)	
Rh-105	2.13 (12)	2.12 (12)	2.12 (12)	1.89 (12)	1.53 (12)	6.20 (11)	
Ru-106	2.87 (10)	2.86 (10)	2.86 (10)	2.86 (10)	2.86 (10)	2.85 (10)	2.41 (10)
Rh-106	2.87 (10)	2.86 (10)	2.86 (10)	2.86 (10)	2.86 (10)	2.85 (10)	2.41 (10)
Tc-107	3.79 (11)	3.62 (5)					
Ru-107	4.73 (11)	3.87 (9)	2.13 (7)				
Rh-107	4.73 (11)	2.63 (11)	1.19 (11)	2.98 (3)			
Ru-108	1.89 (11)	1.04 (9)	5.80 (6)				
Rh-108	1.89 (11)	1.13 (9)	6.27 (6)				
Rh-109	6.63 (10)	4.68 (10)	3.31 (10)	1.62 (7)	3.96 (3)		
Pd-109	6.63 (10)	6.60 (10)	6.53 (10)	3.88 (10)	2.10 (10)	1.82 (9)	
Ag-109m	6.63 (10)	6.60 (10)	6.53 (10)	3.88 (10)	2.10 (10)	1.82 (9)	
Pd-111	4.26 (10)	1.65 (10)	6.43 (9)	6.02 (1)			
Ag-111	3.37 (10)	3.37 (10)	3.36 (10)	3.22 (10)	3.08 (10)	2.56 (10)	8.29 (6)

TABLE C- I (cont.)

ATR FISSION PRODUCT INVENTORY AFTER 17 DAYS OPERATION

Isotope	Disintegrations per Second per Gram of Uranium						
	Time After Shutdown						
	0	1/2 hour	1 hour	12 hours	24 hours	3 days	90 days
Pd-112	2.60 (10)	2.56 (10)	2.51 (10)	1.75 (10)	1.17 (10)	2.42 (9)	
Ag-112	2.60 (10)	2.60 (10)	2.59 (10)	2.03 (10)	1.38 (10)	2.85 (9)	
Ag-113	2.37 (10)	2.21 (10)	2.07 (10)	4.93 (9)	1.02 (9)	1.93 (6)	
Cd-113m	7.36 (5)	7.36 (5)	7.37 (5)	7.46 (5)	7.49 (5)	7.49 (5)	7.25 (5)
Ag-114	2.37 (10)	7.24 (5)	2.21 (1)				
Ag-115	2.60 (10)	9.02 (9)	3.25 (9)				
Cd-115	2.36 (10)	2.40 (10)	2.39 (10)	2.08 (10)	1.78 (10)	9.78 (9)	1.31 (8)
In-115m	2.36 (10)	2.35 (10)	2.35 (10)	2.17 (10)	1.88 (10)	1.01 (10)	
Cd-117m	2.37 (10)	2.10 (10)	1.87 (10)	1.48 (9)	9.25 (7)	1.41 (3)	
Cd-117	2.37 (10)	2.31 (10)	2.20 (10)	2.04 (9)	1.28 (8)	1.95 (3)	
In-117m	2.37 (10)	2.36 (10)	2.35 (10)	5.50 (9)	4.61 (8)	8.34 (3)	
In-117	6.63 (9)	6.62 (9)	6.62 (9)	2.13 (9)	1.95 (8)	3.68 (3)	
Sn-117m	4.64 (7)	4.64 (7)	4.64 (7)	4.64 (7)	4.55 (7)	4.12 (7)	5.56 (5)
Cd-118	2.37 (10)	1.56 (10)	1.03 (10)	1.09 (6)	5.08 (1)		
In-118	2.37 (10)	1.71 (10)	1.32 (10)	1.20 (6)	5.58 (1)		
In-119	2.37 (10)	7.21 (9)	2.19 (9)				
Sn-119m	1.09 (9)	1.08 (9)	1.08 (9)	1.08 (9)	1.08 (9)	1.08 (9)	8.49 (8)
Sn-121	3.31 (10)	3.27 (10)	3.23 (10)	2.44 (10)	1.81 (10)	5.40 (9)	
Sn-123m	2.36 (8)	2.35 (8)	2.35 (8)	2.35 (8)	2.34 (8)	2.32 (8)	1.49 (8)
Sn-123	3.55 (10)	2.09 (10)	1.23 (10)	1.16 (5)			
Sn-125m	2.60 (10)	3.11 (9)	3.73 (8)				
Sn-125	2.03 (10)	2.02 (10)	2.02 (10)	1.95 (10)	1.88 (10)	1.62 (10)	2.66 (7)
Sb-125	6.15 (8)	6.15 (8)	6.15 (8)	6.24 (8)	6.33 (8)	6.65 (8)	8.08 (8)
Te-125m	9.67 (6)	9.69 (6)	9.72 (6)	1.02 (7)	1.08 (7)	1.33 (7)	9.95 (7)
Sn-126	2.37 (11)	1.56 (11)	1.03 (11)	1.09 (7)	5.08 (2)		
Sb-126	2.37 (11)	2.35 (11)	2.31 (11)	1.03 (11)	4.11 (10)	1.02 (9)	
Sn-127	5.68 (11)	4.51 (11)	3.58 (11)	2.22 (10)	8.71 (6)		
Sb-127	5.63 (11)	5.62 (11)	5.62 (11)	5.23 (11)	4.78 (11)	3.34 (11)	5.89 (4)
Te-127m	6.93 (9)	6.63 (9)	6.64 (9)	6.89 (9)	7.15 (9)	7.99 (9)	5.78 (9)
Te-127	4.77 (11)	4.76 (11)	4.76 (11)	4.67 (11)	4.42 (11)	3.19 (11)	5.78 (9)
Sn-128	1.18 (12)	8.21 (11)	5.70 (11)	1.87 (8)	2.95 (4)		
Sb-128	1.18 (12)	7.68 (11)	6.92 (11)	2.28 (8)	3.61 (4)		
Sb-129	2.37 (12)	2.19 (12)	2.03 (12)	3.88 (11)	6.36 (10)	4.60 (7)	
Tc-129m	1.66 (11)	1.66 (11)	1.66 (11)	1.67 (11)	1.65 (11)	1.59 (11)	2.63 (10)
Te-129	1.97 (12)	1.94 (12)	1.89 (12)	5.69 (11)	2.31 (11)	1.59 (11)	2.63 (11)
Sn-130	4.73 (12)	1.59 (9)	5.37 (5)				
Sb-130	4.73 (12)	1.06 (12)	1.89 (11)				
Sb-131	6.87 (12)	2.79 (12)	1.13 (12)	2.85 (3)			
Te-131m	1.03 (12)	1.02 (12)	1.01 (12)	7.90 (11)	5.99 (11)	1.97 (11)	
Te-131	6.83 (12)	5.60 (12)	3.84 (12)	8.01 (11)	6.07 (11)	2.00 (11)	
I-131	5.24 (12)	5.23 (12)	5.23 (12)	5.07 (12)	4.89 (12)	3.17 (12)	2.23 (9)
Xe-131m	2.09 (11)	2.08 (11)	2.08 (11)	2.04 (11)	1.99 (11)	1.82 (11)	1.70 (9)

TABLE C-I (cont.)

ATR FISSION PRODUCT INVENTORY AFTER 17 DAYS OPERATION

Isotope	Disintegrations per Second per Gram of Uranium						
	Time After Shutdown						
	0	1/2 hour	1 hour	12 hours	24 hours	3 days	90 days
Sn-132	8.05 (12)	6.33 (8)	4.98 (4)				
Sb-132	8.05 (12)	4.50 (9)	4.72 (5)				
Te-132	1.02 (13)	1.01 (13)	1.00 (13)	9.11 (12)	8.18 (12)	5.31 (12)	3.66 (4)
I-132	1.02 (13)	1.01 (13)	1.01 (13)	9.40 (12)	8.44 (12)	5.48 (12)	3.78 (4)
Te-133m	1.09 (13)	7.82 (12)	5.62 (12)	3.96 (9)	1.44 (6)		
Te-133	1.42 (13)	8.08 (12)	5.81 (12)	4.09 (9)	1.48 (6)		
I-133	1.54 (13)	1.52 (13)	1.51 (13)	1.06 (13)	7.10 (12)	1.40 (12)	
Xe-133m	3.66 (11)	3.65 (11)	3.65 (11)	3.58 (11)	3.37 (11)	2.20 (11)	
Xe-133	1.34 (13)	1.33 (13)	1.34 (13)	1.33 (13)	1.30 (13)	1.08 (13)	1.21 (8)
So-134	7.10 (12)	1.03 (2)					
Te-134	1.59 (13)	4.12 (12)	1.04 (12)				
I-134	1.80 (13)	1.48 (13)	1.07 (13)	1.83 (9)	1.36 (5)		
Te-135	9.94 (12)	3.04 (8)	9.30 (3)				
I-135	1.40 (13)	1.33 (13)	1.26 (13)	4.03 (12)	1.16 (12)	7.98 (9)	
Xe-135m	4.19 (12)	4.10 (12)	3.93 (12)	1.25 (12)	3.62 (11)	2.49 (9)	
Xe-135	1.54 (13)	1.52 (13)	1.52 (13)	1.06 (13)	5.56 (12)	2.12 (11)	
I-136	1.40 (13)	7.48 (6)	3.76 (0)				
Cs-136	8.80 (9)	8.77 (9)	8.76 (9)	8.55 (9)	8.32 (9)	7.48 (9)	6.98 (7)
I-137	1.16 (13)						
Xe-137	1.33 (13)	6.96 (10)	3.36 (8)				
C-137	1.59 (10)	1.59 (10)	1.59 (10)	1.59 (10)	1.59 (10)	1.59 (10)	1.58 (10)
Ba-137m	1.46 (10)	1.46 (10)	1.46 (10)	1.46 (10)	1.46 (10)	1.46 (10)	1.45 (10)
I-138	8.05 (12)						
Xe-138	1.30 (13)	3.84 (12)	1.13 (12)	2.34 (10)			
Cs-138	1.37 (13)	1.06 (13)	6.61 (12)	7.19 (6)	1.87 (0)		
I-139	4.26 (12)						
Xe-139	1.11 (13)						
Cs-139	1.40 (13)	1.66 (12)	1.86 (11)				
Ba-139	1.42 (13)	1.23 (13)	9.82 (12)	4.55 (10)	1.28 (8)		
Xe-140	8.76 (12)						
Cs-140	1.42 (13)	1.05 (5)					
Ba-140	8.97 (12)	8.96 (12)	8.95 (12)	8.73 (12)	8.50 (12)	7.62 (12)	6.86 (10)
La-140	8.07 (12)	8.07 (12)	8.08 (12)	8.21 (12)	8.29 (12)	8.12 (12)	7.91 (10)
Xe-141	4.26 (12)						
Cs-141	1.11 (13)						
Ba-141	1.40 (13)	4.40 (12)	1.38 (12)	1.28 (11)			
La-141	1.42 (13)	1.36 (13)	1.26 (13)	1.63 (12)	1.72 (11)	2.14 (7)	
Ce-141	4.28 (12)	4.28 (12)	4.29 (12)	4.30 (12)	4.09 (12)	6.50 (11)	
Cs-142	8.05 (12)	7.53 (3)					
Ba-142	1.33 (13)	4.65 (11)	1.45 (10)				
La-142	1.40 (13)	1.15 (13)	8.70 (12)	1.81 (10)	2.14 (7)		
Ba-143	1.16 (13)						
La-143	1.47 (13)	5.02 (12)	1.68 (12)	5.92 (1)			
Ce-143	1.47 (13)	1.46 (13)	1.45 (13)	1.15 (13)	8.95 (12)	3.27 (12)	
Pr-143	7.77 (12)	7.77 (12)	7.78 (12)	7.90 (12)	7.96 (12)	7.73 (12)	9.93 (10)

TABLE C-I (cont.)

ATR FISSION PRODUCT INVENTORY AFTER 17 DAYS OPERATION

Isotope	Disintegrations per Second per Gram of Uranium						
	Time After Shutdown						
	0	1/2 hour	1 hour	12 hours	24 hours	3 days	90 days
Ce-144	5.75 (11)	5.75 (11)	5.75 (11)	5.75 (11)	5.73 (11)	5.71 (11)	4.63 (11)
Pr-144	5.75 (11)	5.75 (11)	5.75 (11)	5.75 (11)	5.73 (11)	5.71 (11)	4.63 (11)
Ce-145	9.94 (12)	9.72 (9)	9.51 (6)				
Pr-145	9.94 (12)	9.46 (12)	8.92 (12)	2.48 (12)	6.13 (11)	2.29 (9)	
Ce-146	7.56 (12)	1.70 (12)	3.80 (11)				
Pr-146	7.81 (12)	5.36 (12)	2.74 (12)	2.35 (4)			
Nd-147	3.99 (12)	3.98 (12)	3.98 (12)	3.86 (12)	3.75 (12)	3.32 (12)	1.60 (10)
Pm-147	2.90 (10)	2.90 (10)	2.90 (10)	3.04 (10)	3.17 (10)	3.68 (10)	7.20 (10)
Nd-149	3.08 (12)	2.59 (12)	2.18 (12)	4.81 (10)	7.53 (8)	4.50 (1)	
Pm-149	3.06 (12)	3.06 (12)	3.06 (12)	2.72 (12)	2.34 (12)	1.26 (12)	2.93
Nd-151	1.14 (12)	2.84 (11)	7.11 (10)				
Pm-151	1.18 (12)	1.18 (12)	1.16 (12)	8.82 (11)	6.52 (11)	1.95 (11)	
Sm-151	4.73 (8)	4.74 (8)	4.74 (8)	4.87 (8)	4.97 (8)	5.16 (8)	5.24 (8)
Sm-153	3.54 (11)	3.52 (11)	3.49 (11)	2.97 (11)	2.49 (11)	1.23 (11)	
Sm-155	7.34 (10)	3.03 (10)	1.25 (10)	4.43 (1)			
Eu-155	1.39 (9)	1.39 (9)	1.39 (9)	1.39 (9)	1.39 (9)	1.25 (9)	
Sm-156	3.08 (10)	2.97 (10)	2.87 (10)	1.34 (10)	5.83 (9)	2.10 (8)	
Eu-156	1.71 (10)	1.71 (10)	1.71 (10)	1.72 (10)	1.70 (10)	1.55 (10)	2.10 (8)

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX D
REACTOR BUILDING LEAK RATE COMPUTATIONS

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX D

REACTOR BUILDING LEAK RATE COMPUTATIONS

1. CONSTANTS

1. "U" Factors: -- Roof = 0.14 Btu/degree F/hr/sq ft
Walls = 0.18 Btu/degree F/hr/sq ft
2. Areas (A): -- Roof = 8800 sq ft
Walls = 12,000 sq ft
3. Building Volume: -- 1,640,000 cu ft (gastight volume)
4. Specific Heat of Air
at Constant Pressure: -- Cp = 0.24 Btu/degree F/lb

2. ASSUMPTIONS

1. Operating Temperature of Building Air = 80°F (at time of accident)
2. Outside Temperature Range: (assumed at constant rate of change)

Condition "A" -- 102°F to 44°F to 102°F in 24-hour cycle.
(Reference 1)

Condition "B" -- 100°F to 62°F to 100°F in 24-hour cycle.
(Reference 1)

3. Outside Barometric Range

Condition "A" -- 24.8 inch Hg to 24.3 inch Hg

Condition "B" -- 24.8 inch Hg to 24.55 inch Hg

4. Wind

Condition "A" -- 11.7 mph for 24 hour (Reference 2)

Condition "B" -- 2.75 mph for 24 hour (Reference 2)

3. FORMULAS

1. $w = \frac{PV}{RT}$, where w = wt of air, lb; P = absolute pressure, lb/sq ft; R = constant, ft-lb/lb/degree; T = absolute temperature, degrees; V = volume, cu ft

2. $Q = UA (t_a - t_b)$, where Q = quantity of heat transmitted due to temperature differential, Btu/hr; t_a = outside air absolute temperature, degrees; t_b = building air absolute temperature, degrees
3. $Q = wC_p \Delta t_b$, where Δt_b = building air temperature change in one hour due to addition of Q amount of heat.
4. $\Delta t_b = \frac{UA}{wC_p} (t_a - t_b)$
5. From AI memo leak rate relationships: (Reference 3)

Hollow Metal Door, tending to close -- $4.37 \Delta P + 22.1 (\Delta P)^{1/2}$ cu ft/min

Hollow Metal Door, tending to open -- $4.93 \Delta P + 27.4 (\Delta P)^{1/2}$ cu ft/min

Quick Closing Door, as-received -- $0.0714 \Delta P + 0.426 (\Delta P)^{1/2}$ cu ft/min
where ΔP = Pressure differential, inches of water.

4. COMPUTATIONS

1. Effect of Temperature

$$w = \frac{PV}{RT} = \frac{12.2 \times 144 \times 1,640,000}{53.3 \times 540} = 100,000 \text{ lb}$$

$$UA \text{ (total)} = 0.14 \times 8800 + 0.18 \times 12,000 = 1230 + 2160 = 3390 \text{ Btu/degree/hr}$$

$$\Delta t_b = \frac{UA}{wC_p} (t_a - t_b) = \frac{3390}{100,000 \times 0.24} (t_a - t_b) = 0.1413 (t_a - t_b)$$

Condition "A": due to range of 23.87° between 61.31 and 85.18° ,

$$V_{\text{diff}} = 1,640,000 \left(\frac{545.18}{521.31} - 1 \right) = 75,112 \text{ cu ft say } 75,000 \text{ cu ft}$$

Condition "B": due to range of 15.74° between 72.83° and 88.51° ,

$$V_{\text{diff}} = 1,640,000 \left(\frac{548.51}{532.83} - 1 \right) = 48,216 \text{ cu ft say } 49,000 \text{ cu ft}$$

Table D-1 shows the range of Δt_b due to the outside temperature change.

2. Effect of Barometer

Condition "A": due to barometric drop of $1/2$ inch Hg,

$$V_{\text{diff}} = 1,640,000 \left(\frac{24.8}{24.3} - 1 \right) = 34,000 \text{ cu ft}$$

Condition "B": due to barometric drop of $1/4$ inch Hg,

$$V_{\text{diff}} = 1,640,000 \left(\frac{24.8}{24.55} - 1 \right) = 17,000 \text{ cu ft}$$

3. Effect of Wind

The south wall has three doors and 150 linear feet of joint between liner and masonry or concrete. Doors are taken as equivalent to "Hollow Metal Doors,"

TABLE D-I
REACTOR BUILDING LEAK RATE COMPUTATIONS

Time (hours)	t_a		t_b		$t_a - t_b$		Δt_b		
	A	B	A	B	A	B	A	B	
1	99.60	98.42	80	80	19.60	18.42	+2.77	+2.60	
2	94.76	95.25	82.77	82.60	11.99	12.65	+1.70	+1.79	
3	89.93	92.07	84.47	84.39	5.46	7.68	+.71	+1.09	
4	85.10	88.90	<u>85.18</u>	85.48	-.08	3.42	-.01	+.48	Maximum Condition "A"
5	80.26	85.73	85.17	85.96	-3.91	-.23	-.65	-.03	Temperature Range Condition "A" 23.87°
6	75.44	82.56	84.52	85.93	-9.08	-3.37	-1.28	-.48	
7	70.61	79.40	83.24	85.45	-12.63	-6.05	-1.79	-.85	
8	65.78	76.22	81.45	84.60	-15.67	-8.38	-2.22	-1.28	
9	60.95	73.05	79.23	83.32	-18.28	-10.27	-2.56	-1.45	
10	56.12	69.89	76.67	81.87	-20.55	-11.98	-2.90	-1.69	
11	51.29	66.72	73.77	80.18	-22.48	-13.46	-3.18	-1.90	
12	46.46	63.54	70.59	78.28	-24.13	-14.74	-3.41	-2.08	
13	46.46	63.54	67.18	76.20	-20.74	-12.66	-2.93	-1.79	
14	51.29	66.72	64.25	74.41	-12.96	-7.69	-1.83	-1.09	
15	56.12	69.89	62.42	73.32	-6.30	-3.43	-.89	-.49	
16	60.95	73.05	61.53	<u>72.83</u>	-1.58	.22	-.22	+.03	Minimum Condition "B"
17	65.78	76.22	<u>61.31</u>	72.86	+4.47	+3.36	+.63	+.47	Minimum Condition "A"
18	70.61	79.40	61.94	73.33	+8.67	+6.07	+1.23	+.86	Temperature Range Condition "B" 15.74°
19	75.44	82.56	63.17	74.19	+12.27	+8.37	+1.74	+1.19	
20	80.26	85.73	64.91	75.38	+15.35	+10.35	+2.17	+1.47	
21	85.10	88.90	67.08	76.85	+18.02	+12.05	+2.55	+1.70	
22	89.93	92.07	69.63	78.55	+20.30	+13.52	+2.87	+1.91	
23	94.76	95.25	72.50	80.46	+22.26	+14.79	+3.15	+2.09	
24	99.60	98.42	75.65	82.55	+23.95	+15.87	+3.39	+2.24	
25	99.60	98.42	79.04	84.79	+20.56	+13.63	+2.91	+1.93	
26	94.76	95.25	81.95	86.72	+12.81	+8.53	+1.81	+1.21	
27	89.93	92.07	83.76	87.93	+6.17	+4.14	+.87	+.58	
28	85.10	88.90	84.63	88.51	+.47	+.39	+.07	+.06	
29	80.26	85.73	84.70	<u>88.57</u>					Maximum Condition "B"

tending to close", and the joint is taken as equivalent to "Quick-Opening Doors, as-received". On this basis, leak rates are computed as follows:

Condition "A": due to 11.7 mph wind

$$P = 0.0033 V^2 = 0.45 \text{ lb/sq ft}$$

$$\Delta P = \frac{0.45}{144} \times 27.7 = 0.087 \text{ inch of water; } (\Delta P)^{1/2} = 0.295$$

$$4.37 \times 0.087 + 22.1 \times 0.295 = 0.38 + 6.52 = 6.9 \text{ cu ft/min}$$

$$0.0714 \times 0.087 + 0.426 \times 0.295 = 0.006 + 0.125 = 0.13 \text{ cu ft/min}$$

$$3 \times 6.9 \times 60 \times 24 = 29,800$$

$$\text{Equivalent doors} = \frac{150}{15} = 10$$

$$10 \times 0.13 \times 60 \times 24 = \underline{1,900}$$

$$\text{Total Leakage} = \underline{31,700} \text{ cu ft.}$$

Condition "B": Due to 2.75 mph wind

$$P = 0.0033 V^2 = 0.025 \text{ lb/sq ft}$$

$$\Delta P = \frac{0.025}{144} \times 27.7 = 0.0048 \text{ inch of water; } (\Delta P)^{1/2} = 0.0693$$

$$4.37 \times 0.0048 + 22.1 \times 0.0693 = 0.021 + 1.53 = 1.55 \text{ cu ft/min}$$

$$0.0714 \times 0.0048 + 0.426 \times 0.0693 = 0.00035 + 0.295 = 0.02985 \text{ cu ft/min}$$

$$3 \times 1.55 \times 60 \times 24 = 6700$$

$$10 \times 0.03 \times 60 \times 25 = \underline{430}$$

$$\text{Total leakage} = 7130 \text{ cu ft.}$$

4. Effect of Reactor Accident

Most critical condition for leakage is release of water from Primary System into Basement. Net releases are as follows:

First 30 minutes	--	7,500 cu ft
Second 30 minutes	--	7,000 cu ft
Next 23 hours	--	<u>22,000</u> cu ft
Total Release	--	36,500 cu ft (24 hours).

5. Combined Leak Rate With All Factors Acting Concurrently

	<u>Condition "A"</u>	<u>Condition "B"</u>
Temperature	75,000 cu ft	49,000 cu ft
Barometer	34,000 cu ft	17,000 cu ft
Wind	31,700 cu ft	7,200 cu ft
Accident	<u>36,500</u> cu ft	<u>36,500</u> cu ft
Total Leakage/24 Hours	<u>177,200</u> cu ft	<u>109,700</u> cu ft
Building Leak Rate/24 Hours (1st)	<u>10.75</u> percent	<u>6.7</u> percent
Building Leak Rate/24 Hours (2nd on)	<u>8.6</u> percent	<u>4.5</u> percent

6. Effect of All Factors on Building Pressure Rise Assuming All Are Acting at Maximum Rates of Application Concurrently

(a) Due to constant temperature rise of 24° in 12 hours, leak rate of $\frac{75,000}{12}$ of 6250 cu ft/hr is required to equalize pressure.

(b) Due to barometer drop of 1/2 inch Hg in three hours, leak rate of $\frac{34,000}{3}$ or 11,330 cu ft/hr is required to equalize pressure.

(c) Since the wind has been assumed to act steadily for the full 24 hours, the full velocity pressure of 0.45 lb/sq ft on the south wall will be added to resultant of other factors.

The possibility of the above three factors occurring simultaneously is remote.

(d) Due to reactor accident, leak rate of 15,000 cu ft/hr is required to equalize pressure.

(e) Total leak rate required through all building penetrations to prevent pressure buildup in building will be:

$$6250 + 11,330 + 15,000 = 32,580 \text{ cu ft/hr} = 542 \text{ cfm}$$

Penetrations subject to leakage are:

28 doors or hatches equivalent to "H. M. Door, tending to close"

9 doors or hatches equivalent to "H. M. Doors, tending to open"

1800 linear feet of Miscellaneous penetrations, equivalent to "Quick-acting Door, as-received", Proration of the door perimeter of 14.17 linear feet to the total perimeter gives the equivalent of 125 "Quick-acting Doors".

Equating the leak rate relationship gives:

$$28(4.37 \Delta P + 22.1 \Delta P^{1/2}) + 9(4.93 \Delta P + 27.4 \Delta P^{1/2}) + 125(0.0714 \Delta P + 0.426 \Delta P^{1/2}) = 542$$

$$122.2 \Delta P + 618 \Delta P^{1/2} + 44.4 \Delta P + 247 \Delta P^{1/2} + 8.9 \Delta P + 53.3 \Delta P^{1/2} = 542$$

$$175.5 \Delta P + 918.3 \Delta P^{1/2} = 542$$

$$\Delta P + 5.23 \Delta P^{1/2} - 3.66 = 0$$

$$(\Delta P^{1/2} + 5.855) (\Delta P^{1/2} - 0.625) = 0$$

$$\Delta P^{1/2} = 0.625$$

$$\Delta P = 0.391 \text{ inch water}$$

$$\frac{0.391}{27.7} = 0.0141 \text{ psi} = 2.03 \text{ lb/sq ft}$$

Adding wind pressure - 0.45 lb/sq ft

∴ maximum pressure on wall = 2.48 lb/sq ft = 0.0174 psi = 0.476 inch H₂O

Approximately maximum stress on wall steel = $\frac{0.0174}{0.15} \times 20,000 = 2320$ psi
(plate) or 2550 psi (struct)

or $\frac{2320}{32,000} = 7.05$ percent of yield strength (plate) or 6.55 percent (struct).

7. Effect of Shutoff Head of H and V Supply System

At shutoff head of 6 inches H₂O, exhaust system will release all of maximum fluid release. Thus, maximum pressure in building will be 6 inches H₂O = 0.216 psi.

Approximate maximum stress on wall steel = $\frac{0.216}{0.15} \times 20,000 = 28,800$ psi
(plate)

$\frac{28,800}{33,000} = 87.5$ percent of yield strength (plate)

$\frac{0.216}{0.15} \times 22,000 = 31,700$ psi (struct)

$\frac{31,700}{39,000} = 81.3$ percent of yield strength (struct).

Members are not stressed to full allowable amount; therefore, it is safe to say that stress in strut frame will not exceed 80 percent of yield strength.

5. REFERENCES

1. G. A. DeMarrais, The Engineering Climatology of the National Reactor Testing Station, United States Weather Bureau, Idaho Falls, Idaho, IDO-12004 (November 1958).
2. G. A. DeMarrais and N. F. Isplitzer, Diffusion Climatology of the National Reactor Testing Station, United States Weather Bureau, Idaho Falls, Idaho, IDO-12015 (April 1960).
3. L. Baurmash et al, Leakage Characteristics of Openings for Reactor Housing Components, Atomics International, NAA-SR-MEMO-5137 (June 20, 1960).

APPENDIX E
RESEARCH AND DEVELOPMENT PROGRAMS

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX E

RESEARCH AND DEVELOPMENT PROGRAMS

A list and brief description of the Research and Development programs associated with the ATR design that have been completed or currently in progress is presented in the appendix.

1. BURNOUT HEAT TRANSFER TESTS

Burnout heat transfer tests were performed at ANL and are currently in progress at Hanford.

1.1 ANL (See IDO-24465) [a]

These tests investigated burnout heat transfer in a partially simulated ATR fuel plate channel. The plates were constructed and heated electrically to simulate a cosine heat generation profile in the axial direction.

1.2 Hanford (Work currently in progress)

These tests are investigating burnout heat transfer as above with the addition of a transverse heat generation profile simulating ATR conditions for the hot channel.

2. HYDRAULIC TESTS

Extensive hydraulic tests were performed in developing the ATR fuel element. These tests ranged from static deflection tests of single plates to dynamic proof tests of the designed element at temperature.

2.1 External Fuel Element Hydraulic Tests (See IDO-24466)

These tests included full scale dynamic testing of various interim ATR fuel element designs and provided information leading to the final element design.

2.2 ATR Fuel Element Hydraulic Proof Tests (Continuing program)

These dynamic tests are providing information on performance of the final design ATR fuel elements as produced by commercial vendors.

2.3 Fuel Element Hydraulic Buckling Tests (See IDO-24463)

These tests provided information to aid in the design of the fuel element.

[a] Distributed under TID-4500

2.4 Fuel Plate Pressure Deflection Tests (See IDO-24464)

These tests provided data to develop analytical techniques to aid in the design of the fuel element.

2.5 Differential Thermal Expansion Tests (See IDO-24461)

The tests demonstrated fuel plate resistance to rippling caused by thermal stresses.

3. PROCEDURES FOR FABRICATING ALUMINUM-BASE ATR FUEL ELEMENTS (See ORNL 3632)^[a]

Procedures were developed for producing commercially acceptable fuel elements.

4. FUEL IRRADIATION TESTS (Continuing program)

ATR fuel performance under irradiation at conditions expected in the ATR is under evaluation. Fuel burnup, corrosion, and metallurgical behavior are being examined in these tests.

5. FUEL ELEMENT CLADDING CORROSION TESTS (To be published [ORNL])

These tests determined the corrosion resistance of several aluminum alloys under simulated ATR conditions and provided information for the final element design.

6. MATERIALS OF CONSTRUCTION TESTS (See IDO-24460)^[a]

These tests examined corrosion rates of the various materials used in the ATR vessel under simulated operating conditions.

7. THERMITE REACTION EVALUATION PROGRAM (See Final Report Project No. B-153 Task II, Reactions in Al-35 Weight Percent U₃O₈ Dispersion, Georgia Institute of Technology, Engineering Experiment Statistics.)

These tests determined thermite reaction rates for ATR fuel material.

^[a] Distributed under TID-4500

8. ATR CRITICAL EXPERIMENT FINAL REPORT (See IDO-24468) [a]

The extensive critical facility experiments provided data for core design, analytical models, and safety analysis.

Additional critical facility tests are continuing in the ATRC.

9. MECHANICAL MOCK-UP TESTS

A mechanical mock-up of the ATR vessel internals provided information as to feasibility of handling and maintenance operations and in developing handling tools.

10. SERVO REGULATOR ROD TESTS (See IDO-24462) [a]

The tests provided performance data to aid in the design and evaluation of the ATR control system.

11. DESIGN STUDIES

Several extensive design studies were made in support of the ATR project. These include:

- (1) Final Shielding Design Report -- IDO-24467 [a]
- (2) TURBO Report -- IDO-24459 [a]
- (3) Closed Loop Analog Study -- IDO-24454 [a]

[a] Distributed under TID-4500

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX F

ERROR PROPAGATION ANALYSIS OF THERMAL BURNOUT

THIS PAGE
WAS INTENTIONALLY
LEFT BLANK

APPENDIX F

ERROR PROPAGATION ANALYSIS OF THERMAL BURNOUT

In the initial thermal burnout studies for ATR fuel channel geometry and operating conditions, it was observed that thermal burnout was obtained whenever the bulk coolant outlet temperature achieved the saturation temperature at the local absolute pressure at the channel exit. The application of this observation together with the concept that there is no transverse mixing of the coolant in the hot channel presents a conservative approach to the thermal burnout analysis. However, the probability of obtaining thermal burnout if the reactor is operated inadvertently at any of the several power reduction setpoints is not indicated without performing a statistical analysis of the circumstances. This appendix presents the statistical analysis that was developed to allow this assessment.

The method used involves the development of an error propagation analysis of those parameters affecting the parameter of interest -- the temperature differential observed in the thermal burnout studies.

$$\text{Let} \quad \phi = T_{\text{sat}} - t_b$$

where

T_{sat} = saturation temperature for the pressure at channel exit, °C

t_b = hot streak temperature at channel exit, °C.

The thermodynamic properties of water-steam were related in the pressure range of 200 to 275 psia by a binomial equation:

$$T_{\text{sat}} = 132.5 + 0.3852p - (0.3806 \times 10^{-3}) p^2$$

where

p = absolute pressure, psia.

The pressure at the channel exit is related to the inlet pressure by the following relationship:

$$p_{\text{out}} = p_{\text{in}} - \Delta p$$

where

p_{out} = absolute pressure at channel exit, psia

p_{in} = inlet plenum pressure, psia

$$\Delta p = \frac{\rho V_{\text{HC}}^2}{144 (2g)} \left[k + 1 + \frac{fL}{De} \right]$$

ρ = bulk coolant density, lb/ft³

V_{HC} = velocity in hot channel, fps

g = gravitational constant, ft/sec²

k = coefficient for sudden contraction, dimensionless

f = friction factor, dimensionless

L = channel length, ft

$$De = \frac{4 (wt)}{2 (w + t)}, \text{ ft}$$

w = channel width, ft

t = channel thickness, ft.

The bulk coolant temperature at the channel exit is related to the inlet coolant temperature as follows:

$$t_b = t_{in} + \Delta t_b \text{ and}$$

$$\Delta t_b = P \left(\frac{F_c}{F_T} \right) K^1 / (wt) \rho V_{HC} C_p$$

where

P = reactor power, MW

F_c/F_T = fraction of core power generated in hot channel, dimensionless

C_p = water specific heat, Btu/lb, °F = 1.0

K^1 = conversion factor, 3.41×10^6 Btu/hr, MW.

The maximum bulk water temperature rise is related to the nominal by:

$$\Delta t_{b,max} = \Delta t_{b,nom} (1 + Z_1)(1 + Z_2)(1 + Z_3)$$

where

$(1 + Z_1)$ = radial peaking factor

$(1 + Z_2)$ = deviation in reactor power

$(1 + Z_3)$ = deviation in fuel content.

The velocity in the hot channel is related to the nominal by:

$$V_{HC} = V_{nom} (1 - Z_4)$$

where

$(1 - Z_4)$ = deviation caused by inlet plenum distribution.

The above relationships are combined to yield a relationship for ϕ in terms of parameters that are either controlled during reactor operation or that are measured elsewhere (eg, fuel inspection or hydraulic testing):

$$\phi = 132.5 + 0.3852 \left\{ p_{in} - \frac{\rho V_{HC}^2}{144 (2g)} \left[k + 1 + \frac{fL}{De} \right] \right\} - 0.3806 \times 10^{-3} \left\{ p_{in} - \frac{\rho V_{HC}^2}{144 (2g)} \left[k + 1 + \frac{fL}{De} \right] \right\}^2 - \left\{ t_{in} + \frac{P \left(\frac{F_c}{F_T} \right) (1 + Z_1)(1 + Z_2)(1 + Z_3) K^1}{3600 (1.8)(wt) \rho V_{HC}} \right\}$$

Execution of the error propagation technique consists of taking the partial differential of ϕ with respect to each of the parameters of the above equation, assigning a variance to each parameter, and relating them as follows:

$$V(\phi) = \sum \left(\frac{\partial \phi}{\partial X} \right)^2 V(X)$$

where

$V(\phi)$ = the variance of ϕ

$\frac{\partial \phi}{\partial X}$ = the partial derivative of ϕ with respect to each of the parameters

$V(X)$ = the variance of each of the parameters.

The partial derivatives for the parameters are:

$$\frac{\partial \phi}{\partial p_{in}} = 0.3852 - (0.7612 \times 10^{-3}) p = C$$

$$\frac{\partial \phi}{\partial \rho} = \frac{1}{\rho} \left\{ \Delta t_b - C \Delta P \right\}$$

$$\frac{\partial \phi}{\partial w} = - \frac{C \rho V_{HC}^2 L}{144 (2g) De}$$

$$\frac{\partial \phi}{\partial L} = - \frac{C \rho V_{HC}^2 f}{144 (2g) De}$$

$$\frac{\partial \phi}{\partial w} = \frac{C}{w^2} \left\{ \frac{\rho V_{HC}^2 f L}{144 \times 2 (2g)} - \frac{\Delta p De}{2} \right\} + \frac{\Delta t_{b, \max}}{w} \left\{ 1 + \frac{De}{4w} \right\}$$

$$\frac{\partial \phi}{\partial t} = \frac{C}{t^2} \left\{ \frac{\rho V_{HC}^2 f L}{144 \times 2 (2g)} - \frac{\Delta p De}{2} \right\} + \frac{\Delta t_{b,max}}{t} \left\{ 1 + \frac{De}{4t} \right\}$$

$$\frac{\partial \phi}{\partial V_{nom}} = \frac{1}{V_{nom}} \left\{ \Delta t_b - 2C \Delta p \right\}$$

$$\frac{\partial \phi}{\partial P} = \frac{-\Delta t_{b,max}}{P}$$

$$\frac{\partial \phi}{\partial (F_c/F_T)} = \frac{-\Delta t_{b,max}}{F_c/F_T}$$

$$\frac{\partial \phi}{\partial Z_1} = \frac{-\Delta t_b}{1 + Z_1}$$

$$\frac{\partial \phi}{\partial Z_2} = \frac{-\Delta t_b}{1 + Z_2}$$

$$\frac{\partial \phi}{\partial Z_3} = \frac{-\Delta t_b}{1 + Z_3}$$

$$\frac{\partial \phi}{\partial Z_4} = \left\{ 2C \Delta p - \Delta t_b \right\} \frac{1}{1 - Z_4}$$

Further, the variance of each of the parameters is equal to the square of one-half of the limit of error of the parameter; this equivalence assumes that the limit of error encompasses two standard deviations of that parameter. A 12.5 MW limit of error on the reactor power, for example (five percent of 250 MW), assumes that 95 percent of the time the reactor power will be greater than 237.5 and less than 262.5 MW when the reactor operators believe that the power is actually 250 MW.

Table F-I presents the results of this error propagation analysis for the reactor operating at nominal full power conditions.

TABLE F-I

ERROR PROPAGATION ANALYSIS FOR THERMAL
BURNOUT FOR NOMINAL ATR FULL POWER CONDITIONS

$$\Delta t_{b, \max} = 86.7^{\circ}\text{C}$$

	Value	Limit of Error	V (X)	$\frac{\partial \phi}{\partial X}$	$\left(\frac{\partial \phi}{\partial X}\right)^2 V (X)$
Inlet Pressure	360 psia	10.8 psia	29.2	0.111	0.4
Fluid Density	60 #/cf	0.5 #/cf	6.25×10^{-2}	1.88	0.2
Inlet Coefficient	0.5	0.05	6.25×10^{-4}	-3	N11
Friction Factor	0.019	0.003	2.25×10^{-6}	49.9	N11
Channel Length	49.5 in.	0.5 in.	6.25×10^{-2}	Very Small	N11
Channel Width	3.752 in.	0.01 in.	25.0×10^{-6}	34.4	N11
Channel Thickness	0.078 in.	0.006 in.	9.0×10^{-6}	1630	23.9
Velocity	47.8 fps	3.3 fps	2.72	1.35	5.0
1 - Z ₄	1	0.05	6.25×10^{-4}	101.7	6.5
Inlet Temperature	51.7°C	1.7°C	0.72	1	0.7
Reactor Power	250 MW	12.5 MW	39.1	0.347	4.7
F _C /F _T	0.0017	2.7×10^{-4}	184×10^{-10}	0.511×10^5	48.0
1 + Z ₁	2.66	0.133	44.2×10^{-4}	32.6	4.7
1 + Z ₂	1	0.05	6.25×10^{-4}	86.7	4.7
1 + Z ₃	1	0.05	6.25×10^{-4}	86.7	4.7
					$\sum 123.5$
					$\sigma_{\phi} = \sqrt{123.5} = 11.1^{\circ}\text{C}$
					$N_{\sigma} = \frac{400-286}{20} = 5$

**PHILLIPS
PETROLEUM
COMPANY**

ATOMIC ENERGY DIVISION