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SAFETY ANALYSIS REPORT
ADVANCED TEST REACTOR
Volume 1 of 2

D. R. deBoisblanc
S. Cohen

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ATOMIC ENERGY DIVISION

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VOLUME 1 OF 2

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ASSISTANT MANAGER
ATOMIC ENERGY DIVISION

S. Cohen

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SAFETY ANALYSIS REPORT
ADVANCED TEST REACTOR

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I. INTRODUCTION AND SUMMARY

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- I-2 Principal Characteristics
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I-1. INTRODUCTION

1. SUMMARY DESCRIPTION

This document is the Safety Analysis Report for the Advanced Test Reactor (ATR) located in a northwest extension of the MTR-ETR complex at the National Reactor Testing Station in Idaho.

The ATR is a high flux test reactor providing irradiation space in the lattice for nine loop experiments and additional space in the reflectors for smaller experiments. Several design features incorporated in the ATR to improve experiment performance are unique in test reactor technology. These features are (a) the use of flux trapping to provide high thermal neutron populations in the nine loop facilities and to reduce experiment flux variation during a fuel cycle; (b) incorporation of special shim rod designs to retain axial flux symmetry throughout a fuel cycle; and (c) regional power measurement and control to provide capability for power shifting within limits in the several core lobes and thereby optimize neutron flux for particular experiments.

The ATR is provided with all auxiliary facilities that are necessary to ensure safe operation without reliance on other Test Reactor Area (TRA) facilities. A critical facility, ATRC, is also provided that is a duplicate of the ATR core. This facility will provide data for each ATR fuel and experiment loading to ensure ATR safe operation.

The ATR will be operated under the same philosophy with respect to administrative control and safeguard and training procedures that have been successfully used at the MTR and ETR, except where specific ATR requirements dictate differences.

2. IDENTIFICATION OF CONTRACTORS

The ATR design contractor is Ebasco Services Incorporated with the Babcock and Wilcox Company as nuclear subcontractor. These firms also provide inspection services for fabrication of components and construction. The prime contractor for construction is the Fluor Corporation. The ATR will be operated by Phillips Petroleum Company in conjunction with its operation of the MTR, ETR, and associated facilities.

I-2. PRINCIPAL CHARACTERISTICS

1. REACTOR BUILDING

The ATR building is approximately 200 feet square and extends 60 feet above and below grade. Two full basements are provided, housing experiment cubicles and control panels, subpile room, control rod access room, reactor primary system equipment, and ventilation equipment. The building substructure is of heavily reinforced concrete construction supported by subsurface rock. The building superstructure is of steel frame construction with exterior walls primarily of insulated aluminum sandwich panels. A major portion of the building is designed as a gastight confinement volume with a leak rate, after securing ventilation systems, of less than 10 percent per day. The confinement volume includes the reactor operating area, the control room, and the basements. The perimeter surfaces of the confinement volume are of concrete, masonry, or welded steel plate with all penetrations, including doors, hatches, and ventilation ducts carefully sealed or gasketed.

Individual ventilation systems provide separate service to specific areas and are designed so that air flow is from clean to potentially contaminated areas. The ventilation to the gastight confinement volume is interlocked with the radiation monitoring system so that the supply will be shut off on detection of high radioactivity in the reactor room. The exhaust is shut off on detection of high radioactivity at the stack inlet.

2. REACTOR SYSTEM, SHIELDING, AND EXPERIMENTS

The reactor vessel, fabricated of low carbon stainless steel, is approximately 12 feet in diameter and 35 feet in overall height. Internal thermal shields are provided to reduce gamma heating in the concrete biological shield and vessel wall. The vessel is provided with inlet and outlet nozzles for the primary coolant and various penetrations for experiments, instruments, and reactor nuclear detectors. A drop tube attached to the vessel side is used to transfer fuel, certain experiments, and small reactor internals from the vessel directly to an adjacent stainless steel lined canal system. The canal system is used for underwater storage and for disassembly of fuel elements and other radioactive reactor and experiment components. Transfer from the reactor to the canal is performed under water using long handled tools manipulated from above the reactor through five large oval ports in the reactor top head. The full opening top head is removable. Samples from the loop facilities are transferred to a special cask positioned above the loop top-head penetration. The ATRC is located in the Canal System.

ATR shielding is designed for a maximum dose rate of 0.25 mrem/hour in full-time occupancy locations. The principal reactor biological shield consists of ordinary concrete supplemented by three feet of high density concrete opposite the core region, and special inserts of lead, steel, and polyethylene in certain locations. The shielding walls of the experiment cubicles are primarily high density concrete. The canal walls and floor are ordinary concrete.

Six high pressure water loops, essentially identical in design, are scheduled for early insertion in the ATR flux trap positions. In addition, a high temperature gas loop is under design for insertion in a flux trap position. The loops each consist of an in-pile section, out-of-pile equipment housed in a shielded cubicle in the basement, and a control system with safety circuits tied in to reactor safety system.

3. CORE

The ATR core consists of 40 fuel elements arranged in a serpentine annulus between and around the nine flux traps. Each element consists of 19 parallel curved plates of different width, attached to side plates forming a 45 degree sector of a circular annulus in cross section. The plates are fabricated of fully enriched U_3O_8 dispersed in and clad with an aluminum alloy. Boron carbide is uniformly dispersed in the fuel as a burnable poison. The serpentine annular core and flux traps are surrounded by a metallic beryllium reflector.

The reactor is provided with three types of control rods:

(1) Outer shim cylinders are rotating reflector-absorber cylinders located in the beryllium reflector. Each one of 16 cylinders has a portion of its periphery covered by a hafnium absorber plate. Cylinder rotation moves the absorber plate laterally toward or away from the core. The cylinders are driven in pairs through a worm gear by drives located in an access area outside the principal biological shield. These control elements are equipped with interlocks to automatically maintain the regulating rod in the desired range.

(2) Neck shim rods are small, hollow, vertical, hafnium poison rods with aluminum followers. These rods are usually fully inserted or withdrawn. There are six neck shims in each of four neck regions. Each of the 24 rods with the exception of one in each neck is driven by a flexible cable. The other four are driven by rigid rods. The drives are located in a rod-access room below the subpile room. Two of the four rigid drives are equipped with servo systems for automatic regulating action.

(3) The safety rods are hollow cylinders, containing hafnium inserts, concentric with and inside the flux traps. The five rods will initially surround the pressure tube assemblies of five high pressure water loops. In starting up the ATR, the safety rods are fully withdrawn above the core. The safety rods are raised by a rack and pinion drive through a horizontal shaft below the core. The drive units are located in an access area outside the principal biological shield. The safety rods are scrambled by interrupting the current to holding magnets on quick release clutches attached to the pinion drive shafts. When released, the rods are driven downward into the core by gravity and scram spring motors on the drive shaft supplemented by hydraulic force when primary coolant pumps are operating.

4. PRIMARY COOLANT SYSTEM

The reactor is cooled and moderated by a downward flow of demineralized water through the fuel, flux traps, reflector, and control elements. The primary coolant leaving the reactor transfers heat to a secondary water coolant system in four heat exchangers. The secondary coolant pumps heat to the atmosphere in a cooling tower. Four primary coolant pumps (one spare) provide adequate circulation of primary coolant. Two emergency coolant pumps are available to remove core decay heat during emergencies and during shutdown and low power flux runs.

5. INSTRUMENTATION AND ELECTRICAL SYSTEMS

The ATR is provided with nuclear, process, and radiation monitoring instrumentation to permit safe reactor operation under normal and abnormal conditions. Automatic control of reactor power is provided and automatic corrective and safety actions prevent hazardous conditions from developing.

Several independent electrical systems are provided for the ATR to ensure that power sources are available under any credible condition.

These systems are described in detail in Sections VII and IX.

6. AUXILIARY SYSTEMS

The ATR is provided with auxiliary systems allowing operation that is independent of MTR-ETR operation. These systems include demineralizers, steam, air, diesel fuel, and fire water.

7. DESIGN AND EXPECTED PERFORMANCE CHARACTERISTICS

The design characteristics of the ATR are listed in Table I-2-1. Expected performance characteristics of the ATR are listed in Table I-2-2.

TABLE I-2-1

DESIGN CHARACTERISTICS

Reactor thermal power	250 MW
Average core power density	0.96 MW/liter
<u>Fuel Elements</u>	
Type	Parallel curved plates
Number of plates per element	19
Fuel material	U ₃ O ₈ dispersed in X8001 Al
Nominal U ₃ O ₈ concentration	41% by weight
B ₄ C concentration	0.19% by weight
Fuel cladding	6061 O aluminum
Meat thickness	0.020 inches
Clad thickness	
Plates 2-18	0.015 inches
Plate 1	0.030
Plate 19	0.040
Coolant gap between plates	0.078 inch
Active fuel length	48 inches
Side plate	6061 T6 aluminum
Number of elements in core	40
Core loading	39 kg U-235
Metal-to-water ratio	0.8
Coolant-moderator	Demineralized water
Heat transfer area per element	33.8 ft ²
<u>Reflector</u>	
Material	Beryllium metal
Height	51 inches

TABLE I-2-1 (cont.)

DESIGN CHARACTERISTICS

<u>Control Elements</u>	
Outer shims	
Location	Beryllium reflector
Configuration	Beryllium cylinders with 120° poison inserts
Poison inserts	1/4-inch-thick hafnium
Active length	46 inches
Number of cylinders	16 (4 per lobe)
Number of drives	8 (2 per lobe)
Drive type	Right angle worm and wheel
Neck shims and regulating rods	
Location	Necks of outer lobes (4)
Poison section	48-inch-long fluted hafnium hollow rods -- 3/8 inch diam- eter
Follower section	48 x 1/4-inch diameter alumi- num rod
Number of rods	24 (6 per neck)
Number of drives	24
Drive type	Flexible cables (20) Rack and pinion (4) -- 2 which will serve as regulating rods
Safety rods	
Location	Flux traps surrounding five high-pressure water loop as- semblies
Poison section	4 - 90° hafnium sector inserts
Length	36 inches
Follower section	4 - 90° aluminum sector inserts of proper thickness to con- trol flux trap metal-water ratio
Stroke	36 inches
Number of rods	5
Drive type	Rack and pinion with quick release magnetic clutch

TABLE I-2-1 (cont.)

DESIGN CHARACTERISTICS

<u>Reactivity Balance and Control</u>	
Maximum cold clean reactivity $\rho = \frac{k_{eff} - 1}{k_{eff}}$	0.170
Burnable poison (B-10) start of cycle	-0.074
Power effect (sum of positive effect of loops and negative effect of core)	+0.006
Maximum hot, no moving control, start of cycle	0.102
Available movable control	
Outer shim cylinders	0.061
Neck shims	0.048
Shim interaction	<u>0.030</u>
Total	0.139
Safety rods (5)	0.068-0.075
Safety rods (4)	0.046-0.051
<u>Reactor Hydraulics and Thermal Design</u>	
Operating pressure (reactor inlet)	355 psig
Coolant flow -- nominal	47,000 gpm
Fuel element flow	27,400 gpm
Velocity in fuel channels	47.8 fps
Pressure drop across core	100 psi
Reactor inlet coolant temperature	125°F
Reactor outlet coolant temperature -- average	167°F
Reactor outlet coolant temperature -- hot track	356°F
<u>Reactor Vessel</u>	
Material	304 stainless steel (<0.06 carbon)
Design pressure	390 psig
Design temperature	240°F

TABLE I-2-1 (cont.)

DESIGN CHARACTERISTICS

<u>Reactor Vessel</u>	
Outside diameter	12 feet
Overall height	35 feet
Inlet nozzles	Two 23-inch ID
Outlet nozzles	Four 15-inch ID
<u>Primary Coolant System</u>	
Heat exchangers	
Number	4
Tube material	Inconel
Shell material	Carbon steel
Primary coolant pumps	
Number	4
Type	Horizontal, centrifugal, double suction
Flow (each)	18,000 gpm
Emergency coolant pumps	
Number	2
Flow (each)	4,700 gpm
Piping material	304 and 304 L stainless steel
Surge tank	
Total volume	1,000 cubic feet
Operating gas volume	50 cubic feet
Secondary coolant system	
Flow range	2,000 to 31,200 gpm
Pumps	Four 8,600 gpm each
Piping material	Carbon steel
Cooling tower capability	9.76×10^8 Btu/hr

TABLE I-2-2

EXPECTED PERFORMANCE CHARACTERISTICS

Full power maximum lobe power shift	40-50-60 MW
Radial peak/average power in core	2.66
Axial peak/average power in core	1.42
Typical perturbed fluxes for fueled samples in high power flux trap	
Midplane average thermal neutron flux	7.7×10^{14} n/cm ² sec
Midplane average epithermal neutron flux	$15-16 \times 10^{14}$ n/cm ² sec
Core average thermal flux	
Beginning of cycle	2.1×10^{14} n/cm ² sec
End of cycle	2.5×10^{14} n/cm ² sec
Core average epithermal flux	
Beginning of cycle	14.6×10^{14} n/cm ² sec
End of cycle	14.8×10^{14} n/cm ² sec
Fuel cycle	17 days
<u>Reactivity Coefficients</u>	
Fuel region temperature coefficient - $\Delta\rho/^{\circ}\text{F}$	
Beginning of cycle	-1.37×10^{-4}
End of cycle	-0.97×10^{-4}
Fuel region void coefficient - $\Delta\rho/\Delta V/V$	
Beginning of cycle	-0.209
End of cycle	-0.103
<u>Fuel Element Thermal Performance</u>	
Average core heat flux	0.59×10^6 Btu/hr-ft ²
Maximum nominal core heat flux	2.23×10^6 Btu/hr-ft ²
Ratio of burnout power level to operating power level	1.25
Burnout margin at trip level	1.04

8. RESEARCH AND DEVELOPMENT PROGRAMS

The ATR design was based on extensive research and development programs and component testing. A list of these programs is shown below.

- (1) Burnout Heat Transfer Tests
- (2) External Fuel Element Hydraulics Tests
- (3) Fuel Fabrication
- (4) Fuel Element Hydraulic Buckling Tests
- (5) Fuel Element Pressure Deflection Tests
- (6) Fuel Irradiation Tests
- (7) Fuel Element Cladding Corrosion Tests
- (8) Fuel Element Thermal Distortion Tests
- (9) Materials of Construction Tests
- (10) Thermite Reaction Tests
- (11) Critical Experiments
- (12) Control System Tests

Component development testing included extensive work involving the drives of the safety rods, outer shim rods, neck shim rods, and safety rod magnetic clutch.

I-3. SAFETY PRINCIPLES

1. OPERATION AND ADMINISTRATIVE CONTROL

Operation of the ATR will conform to established practices used in the operation of the MTR and ETR. The operations staff consists of Phillips AED senior management members and engineers with broad test reactor operating experience.

The training programs required of all operations personnel include both theoretical and ATR orientated courses. Review and certification by a senior committee is necessary before personnel can operate the reactor. Technical assistance from other branches of Phillips AED is supplied to establish the post neutron start-up program in conjunction with the ATR Start-up Committee.

The technical staff maintains a continuing surveillance review of ATR performance to prevent any hazardous situation from developing. Safeguard review procedures require both approval of the Reactor Safeguards Committee and the Assistant Manager of Phillips AED responsible for test reactor surveillance, prior to making any change that could conceivably affect safety. The Reactor Safeguards Committee also reviews and approves all experiment insertions and withdrawal procedures.

The policies and procedures developed by Phillips AED over many years of test reactor operation continually stress the importance of safe working attitudes and habits above all other considerations. This attitude among all personnel provides the primary defense against the development of hazardous conditions.

2. ATR SAFETY FEATURES

The ATR incorporates the following features in the design to ensure the overall facility safety.

(1) The reactor has a completely independent safety system, separate from the normal operational control system. This safety system includes special instruments, special relay and electronic scram buses, and safety rods. These safety devices are not used for any operational function, and are always available for emergency reduction of reactor power. This safety system works on the auctioneered principle where a single instrument signal directs the safety action.

(2) The heat transfer characteristics of the core provide an adequate margin of safety to prevent local boiling and fuel element burnout.

- (3) The reactor possesses both a negative temperature coefficient and a negative void coefficient in the fuel region, which provide an inherent limitation to reactivity excursions.
- (4) A core tank and elevated exit nozzles inside the reactor vessel provide a continuous envelope that keeps the core covered with cooling water. The exit nozzles have siphon breakers.
- (5) The pressure vessel and primary system components are designed to meet or exceed all applicable codes including all nuclear cases thereof.
- (6) A large amount of water shielding exists between the reactor and the reactor vessel wall, which minimizes fast neutron damage to the reactor vessel.
- (7) The reactor safety system is designed on the basis of a back-up scram philosophy. For each process or control system failure that could lead to fuel melting, there are at least two independent instrumentation detection, safety system circuits to cause the scram. These circuits have as their only common components the process itself and the scram bus. Failure of any component in either circuit does not impair the operation of the other circuit.
- (8) The shim and safety rod drives are designed to limit withdrawal rates to values corresponding to reactivity rates of less than 0.1 percent $\Delta k/\text{sec}$.
- (9) The total reactivity available to the automatic control system and the servo mechanism is limited to less than the delayed neutron fraction.
- (10) The reactor control system and instrumentation are similar to those used in MTR and ETR for which reliability has been demonstrated.
- (11) The reactor has movable control rod worth sufficient for a large shutdown margin. Several shim control elements must be withdrawn to obtain criticality. The safety rods must be at their upper limit switches before shim rods can be withdrawn.
- (12) Uninterrupted flow of primary coolant is assured by two emergency coolant pumps with independent power sources. The minimum coolant flow requirements for removing decay heat can be met after simultaneous loss of both commercial electric power and diesel power.
- (13) Fire water is automatically injected for emergency cooling of the reactor in the event of uncontrolled loss of coolant. This permits cooling the reactor in the shutdown condition indefinitely without a source of electric power.

(14) There are two independent sources of ac electric power, the commercial electric power and a separate set of diesel generators. The important plant loads are divided between these two power sources.

(15) The commercial electrical power distribution system is arranged to provide power distribution through two separate systems. Relaying is provided to isolate one system in case of failure and, thereby, prevent loss of power to the other system. The primary pump motors are divided equally between the two systems. Other multiple equipment motors are similarly divided between the two systems.

(16) A battery-backed emergency power source is available for inserting control rods.

(17) A major portion of the building is designed as a gastight confinement volume with a leak rate, after securing the ventilation system, of less than 10 percent of the confined volume in 24 hours. The confinement volume includes the reactor operating room, the control room, and the basements. The perimeter surfaces of the confinement volume are of concrete, masonry, or welded steel plate with all penetrations including doors, hatches, and ventilation ducts carefully sealed or gasketed.

(18) Building sprays, which significantly lower the iodine and solid fission product concentrations, are provided to reduce the environmental hazards following a release of radioactive material within the reactor building.

(19) A radiation monitoring system provides alarms and indication of abnormal radiation levels to the operating personnel. An automatic ventilation control system, actuated by the plant radiation monitors, prevents the discharge of highly radioactive materials to the stack.

(20) The plant has a waste disposal system to control the discharge of liquid and solid plant effluents and to maintain acceptable release rates of radioactive materials.

(21) The location of the ATR stack is optimized to reduce the effects of radioactive gas release on the adjacent facilities.

(22) Other features such as margins in capacity of all cooling circuits, standby equipment for essential services, safety design features widely used in industrial facilities, and others are discussed in detail throughout this report.

I-4. SUMMARY

The safety analysis is summarized as follows:

1. NORMAL OPERATION

(1) The ATR is designed to assure controlled conditions throughout all modes of normal operation.

(2) The reactor system is designed to provide adequate shielding for all operating personnel during normal conditions.

(3) Gaseous activity from the ATR stack is expected to be less than from either MTR or ETR. In addition, the more favorable location of the ATR stack in the MTR-ETR-ATR complex further reduces the potential effect of the ATR discharge.

(4) Adequate liquid and solid waste disposal for the ATR is provided by incorporation with existing MTR-ETR waste disposal systems.

(5) Operation of the ATR in close proximity to the MTR and ETR will not result in any undesirable coupling effects.

2. ABNORMAL REACTIVITY ADDITIONS

(1) Neither a start-up accident nor a rod withdrawal accident in the power range nor a cold water accident will produce an excursion which will result in fuel element damage. Some fuel melting can occur as a result of the maximum credible reactivity accident.

(2) The regions of the ATR core possess a high degree of nuclear coupling. The power in each lobe is so closely related to the others that safety rod insertion into any core region provides adequate shutdown capability.

(3) Calculations show that the reactor vessel will not rupture as a result of energy releases in the microsecond range below 235 MW seconds. An energy release between 350 and 1200 MW seconds in the millisecond range can be tolerated without vessel rupture. A safety margin of greater than 2 exists for the maximum credible energy release from an ATR excursion.

3. OTHER ABNORMAL INCIDENTS

(1) Loss of any combination of main primary coolant pumps cannot result in fuel element damage.

(2) Loss of primary coolant will not cause melting of fuel up to a rupture size of three inches in diameter.

4. ENVIRONMENTAL HAZARDS

(1) The design of the gastight confinement volume of the ATR reactor building is capable of limiting air leakage to less than 10 percent of the confined volume in 24 hours under maximum anticipated fluctuations in meteorological conditions. The maximum pressure rise resulting from either a rupture of the reactor primary coolant system or a rupture of a pressurized water loop would neither impair the integrity of the confinement volume nor result in an average leak rate greater than 10 percent in 24 hours. Potential missiles resulting from primary system or loop failures would not penetrate the confinement surface.

(2) The environmental hazards analysis was based on:

(a) Complete melting of the core, conservatively selected as the maximum hypothetical accident, which releases 100 percent of the noble gas fission products, 50 percent of the halogen fission products (half of which is absorbed on internal surfaces), and 1 percent of the solid fission products.

(b) Melting of one fuel element which releases 3 percent of the fission products listed for the 100 percent meltdown.

(3) Personnel at MTR-ETR-ATR area and the rest of the NRTS can be evacuated following a complete ATR core meltdown without accumulating an unacceptable integrated dose of either total beta-gamma or iodine.

(4) Calculations indicate that the integrated dose at the largest center of population within 50 miles of the ATR will be less than 300 rem iodine and 25 rem total beta-gamma even with a complete core meltdown. The calculations were based on extremely conservative assumptions.

(5) The use of building sprays significantly reduces the dose to the population off site.

II-5. CONCLUSIONS

(1) The ATR design, if supported by the results of the research and development programs, will meet the required performance criteria.

(2) The ATR design limits the hazard to the environs to an acceptable level.

II. ATR SITE

II-1 Site Description

II-2 References

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II-1. SITE DESCRIPTION

1. LOCATION AND POPULATION

1.1 NRTS Environs

The ATR is located at the National Reactor Testing Station (NRTS), a 900-square-mile tract of land in the semiarid, sparsely populated area of southeastern Idaho. This land, on the Snake River plain at an average elevation of 5000 feet above sea level, is generally flat with a one percent upward slope from southwest to northeast. Mountains rise to an elevation of 10,000 feet along the western boundary of the plain while to the east and southeast the land slopes up approximately 600 feet to the top of a rise running parallel to the Snake River and midway between the site and the river.

The location of the NRTS and its relationship to surrounding populated areas is shown in Figure II-1-1. There are only a few small towns in the immediate vicinity. The closest towns of more than 2000 population, Blackfoot and Shelley, are approximately 40 miles from the ATR site. The closest center of population of more than 8000 is Idaho Falls, 44 miles east of the ATR site.

The distance from ATR and the population of the towns within a 50-mile radius of the ATR site are listed in Table II-1-1.

There are few permanent residences outside of the towns because of the semiarid climate and difficulty in obtaining water supplies. The general area has no permanent streams; a few intermittent streams rise in the mountains and flow into the NRTS disappearing in a number of sinks. Furthermore, the area around NRTS has a considerable fraction of its surface covered by lava flows which make these areas unsuitable for any agricultural use. Accordingly, rural population is concentrated in the Snake River Valley where water is available.

1.2 On NRTS

A number of nuclear reactors and other AEC installations are located on the NRTS as shown on Figure II-1-2.

The ATR is northwest of the existing MTR-ETR area as indicated on Figure II-1-3. The MTR-ETR-ATR area has three main subdivisions: (a) the MTR-ETR area, (b) the ATR area, and (c) a common area for many plant auxiliaries including the electrical substation, the steam plant, the oil storage tanks, the water treatment plant, and the raw water pump house.

The MTR-ETR area contains the Materials Testing Reactor (MTR), the Engineering Test Reactor (ETR), and three zero power reactors: the Advanced Reactivity Measurement Facilities (ARMF) I and II, and the Engineering Test Reactor Critical Facility (ETRC). The ATR reactor building, in the center of the ATR area, is approximately 850 feet northwest of the MTR reactor building and 1200 feet northwest of the ETR reactor building. A new evacuation area will be built at the southern corner of the site outside of the present fence line.

FIG. 11-1-1 MAP OF NATIONAL REACTOR TEST STATION AND ADJACENT AREAS.

TABLE II-1-1

POPULATION DISTRIBUTION AROUND NRTS -- 1960 CENSUS

<u>Location</u>	<u>Distance From ATR</u>	<u>Population</u>	<u>Direction From ATR</u>
Atomic City	10-15 mile zone	141	SE
Howe	10-20 mile zone	20	NNW
Rural Population	10-20 mile zone	301	NNW
Arco-Butte City	20-30 mile zone	1,670	W
Taber	30-40 mile zone	5	SE
Darlington	30-40 mile zone	10	NW
Mud Lake	30-40 mile zone	187	NE
Mackay	30-40 mile zone	650	NW
Monte View	30-40 mile zone	10	NE
Rural Population	30-40 mile zone	991	NE
Moore	30-40 mile zone	358	WNW
Terreton	30-40 mile zone	10	NE
Idaho Falls	40-50 mile zone	33,100	E
Blackfoot	40-50 mile zone	7,378	SE
Shelley	40-50 mile zone	2,612	ESE
Roberts	40-50 mile zone	422	ENE
Firth	40-50 mile zone	322	ESE
Basalt	40-50 mile zone	275	ESE

The facilities at NRTS have a variable working population depending on the status of the programs and the utilization of day or night shifts. The number of personnel at each of the facilities during June 1961, a representative month, is shown in Table II-1-2.

The population at MTR-ETR is expected to increase to 850 and 250 during the day and night, respectively, when the ATR is in operation.

1.3 Estimated Population Growth [4]

The population increase between 1950 and 1960 was not significant for most communities. The increase of about 20,000 inhabitants within 50 miles of the ATR has been mainly in Idaho Falls (increase of 14,000) and in a few other communities such as Arco, Shelley, Ammon, and Blackfoot. Most government and contractor personnel employed at the NRTS reside in Idaho Falls, Blackfoot, Pocatello, or Arco, and the increase in population can be attributed in general to the development of the NRTS. It is expected that future increases in population in this area should be at a slower rate.

FIG. 11-1-3 ATR-MTR-ETR PLOT PLAN.

TABLE II-1-2
PERSONNEL AT NRTS AREAS -- JUNE 1961

<u>Facility</u>	<u>Personnel</u>		<u>Direction From ATR</u>	<u>Distance From ATR (miles)</u>
	<u>Day</u>	<u>Night</u>		
ANP	200	25	NE	22
NRF	1,400	200	NNE	5
MTR-ETR	700	150	SE	--
CPP	300	20	SE	2
Spert	140	5	ESE	5
EBR-I-BORAX	100	2	SSW	5
CFA	800	100	SSE	5
OMRE-EOCR	250	10	SE	5.5
AREA	100	5	ESE	8
EBR-II	200	5	E	15

2. METEOROLOGY

2.1 General

The meteorological conditions at NRTS are described in great detail in a number of reports prepared by the United States Weather Bureau at Idaho Falls, Idaho. Weather records for the region and the NRTS are available from 1921 and 1959, respectively. The latest of the series of reports, IDO-12015 (Reference 1), has been used as the basis of this study.

Records at different observation points at NRTS indicate significant variation in conditions at different sites at a given time. Nevertheless, the observations taken at the Weather Bureau Office (WBO) located at Central Facilities Area (CFA) only five miles from ATR, as reported in IDO-12015 (Reference 1), are considered representative of conditions at ATR.

2.2 Precipitation

The precipitation (rain in summer, snow in winter) at NRTS averages 7.69 inches per year. The greatest fall of rain measured in 24 hours was 1.73 inches during June 1954, and the greatest 24-hour fall of snow was 8.5 inches during January 1957. However, the normal rate of precipitation is small. For example, in May when the total normal rainfall is 1.02 inches, the number of days of precipitation is 13.

2.3 Temperature

The maximum recorded temperature was 102°F on July 18, 1951, and the minimum was minus 35°F on February 1, 1956. Of greater interest is the daily temperature variation range which is between 22 and 25°F in the winter and from 30 to 38°F in the summer.

2.4 Barometric Variations

The normal barometric pressure at the ATR site is 24.8 inches Hg. The greatest 24-hour variation recorded was 0.5 inch Hg, and the greatest 3-hour variation was also 0.5 inch Hg.

2.5 Lapse and Inversion Conditions

Lapse conditions during daylight hours and inversion conditions at night are normal at NRTS. In general, the inversion conditions start at sunset and end shortly after sunrise. The duration of an inversion is closely related to the hours of darkness but may be extended when weather conditions reduce the rate of solar heating. Inversion conditions exist for some period during 92 to 98 percent of the nights of the year and for at least 10 hours on 61 percent of the nights. The frequency of inversions which last more than 10 hours varies with the season of the year. As an example, an inversion lasting 15 hours has a probability of only 0.2 percent in the summer but a 50 percent probability in winter. An inversion lasting 20 hours has a 5 percent probability in the winter and has never been recorded in the summer. However, long duration inversion conditions break up below 250 feet during daylight hours.

Measurements of the thermal gradient at NRTS have been made since 1950. The data on Figures II-1-4 and II-1-5, reproduced from IDO-12015, establish the following basic characteristics (Reference 1):

- (1) Strong inversion conditions corresponding to Sutton's stability constant $n = 0.5$ occur at WBO mainly in the summer months.
- (2) $n = 0.5$ inversions have their greatest duration between 5 and 400 feet during June, July, and August. During these months they occur 3 hours per day. Their average annual occurrence is 25 percent.
- (3) $n = 0.5$ inversions have their greatest duration between 5 and 250 feet during August and September. During these months they occur 8 hours per day. The average annual occurrence is 13 percent.
- (4) Lapse conditions where $n = 0.2$ exist 50 percent of the year.

When the inversions break up there is a period of fumigation. All measurements of the fumigation period indicate that it is of short duration; however, during these periods, stack effluent can reach the ground in relatively undiluted form. The time of fumigation is shown in Figure II-1-6.

2.6 Winds

The wind frequency pattern for NRTS measured at the Weather Bureau Office is shown in Figures II-1-7, -8, -9, -10, -11, and -12. The presence or absence of inversions has an effect on the wind direction and speed. The average wind velocity is lower during inversions than during lapse. However, even during inversions, winds up to 15 miles per hour are fairly common. The winds have higher velocities at greater elevations.

FIG. 11-1-4 WBO AVERAGE THERMAL GRADIENTS BETWEEN 5- AND 250-FOOT LEVELS AND ASSOCIATED N-VALUES, JANUARY 1955 TO MAY 1958.

FIG. 11-1-5 WBO SEASONAL VARIATIONS IN THE AVERAGE THERMAL GRADIENT BETWEEN 5-FOOT AND 250-FOOT LEVELS, SEPTEMBER 1950 THROUGH AUGUST 1952.

FIG. 11-1-6 TIME OF FUMIGATION WBO -- JULY 1951 THROUGH DECEMBER 1953.

The ATR-MTR-ETR area is not under the influence of special wind conditions such as the Birch Creek Valley wind which has been known to blow from the northwest for 60 consecutive hours. Winds blowing from the north and west mountain valleys occur most frequently at night but do not blow into the ATR area.

The winds generally rotate around the compass every day. Winds with north components occur most frequently at night during inversions while winds with south components occur most frequently during the day. There is a time lag of a few hours in the shift of wind direction following a change from lapse to inversion or from inversion to lapse. The wind frequency varies with elevation. The average daily frequency does not exceed 22 percent during inversions or 30 percent during lapse in any direction measured in 30-degree sectors.

2.7 Summary

Analysis of the meteorological data shows that for study of environmental hazards the following are most important.

- (1) The greatest frequency of severe inversions occurs during the summer months.
- (2) During these months, the duration of an inversion is almost never greater than 15 hours per day.
- (3) During these summer inversions, the wind speed is variable and is not persistent in any one direction.

FIG. 11-1-7 CF 250-FOOT LEVEL, WIND ROSES -- JULY 1961 THROUGH MAY 1962,

FIG. 11-1-8 CF 250-FOOT LEVEL, LAPSE WIND ROSES -- JULY 1961 THROUGH MAY 1962.

FIG. 11-1-9 CF 250-FOOT LEVEL, INVERSION WIND ROSES -- JULY 1961 THROUGH MAY 1962.

FIG. 11-1-10 CF 20-FOOT LEVEL, WIND ROSES -- JANUARY 1950 THROUGH MAY 1962.

FIG. 11-1-11 CF 20-FOOT LEVEL, LAPSE WIND ROSES -- JANUARY 1950 THROUGH MAY 1962.

FIG. 11-1-12 CF 20-FOOT LEVEL, INVERSION WIND ROSES -- JANUARY 1950 THROUGH MAY 1962.

3. GEOLOGY AND SEISMOLOGY

3.1 Geology of Site

The Snake River Plain in which the NRTS is located is geologically a broad structural depression which has been filled by extensive accumulation of interlocking basalt flows and layers of lacustrine and alluvial deposits. On top of these layers are considerable amounts of windblown sediments. Much of the area is covered by exposed lava flows of recent geological origin. The geology and hydrology of the site is described in IDO-22033-USGS (Reference 2).

The deposits of permeable volcanic materials such as vesicular lava, pumice, and scoria, over layers of fissured basalt result in a structure which permits surface water to sink down to great depths. Because of the ground structure the NRTS is not crossed by any streams. Streams rising in the mountains to the northwest of the site terminate in sinks. Examples of these are the Big and Little Lost Rivers and Birch Creek.

The MTR-ETR-ATR site is located on the ancient dry flood plain of the Big Lost River. The flood plain deposits consist of coarse sand and gravel with well rounded small cobbles. Clay and silt are present in minor amounts. These deposits are highly permeable and range in thickness from a few feet to more than 60 feet and are underlain by vesicular basalt. Between the basalt and the sand and gravel layers are beds of fine sand and silt where sags occur in the basalt layer. These silt layers have a moderate clay content. The condition of the basalt varies from very dense and massive forms to very vesicular. The site geological structure, with its impermeable layers is of great importance in preventing the pollution of the ground water supply by radioactive effluents.

The northern part of the NRTS contains part of the dried lake bottom of ancient Lake Terreton, but this structure is of no consequence to the MTR-ETR-ATR area.

3.2 Seismology

The Snake River Plain, including NRTS, is subject to occasional seismic tremors. The seismic record for the area is short, the oldest recorded shock having occurred in 1894. In the period 1894 to 1961, there were nineteen quakes in Idaho with an epicenter intensity of V or more on the Rossi-Forel scale. In 1959 near the Madison River Valley in Montana there was a strong quake of intensity X which was felt, but did no damage at NRTS. The NRTS is in earthquake Zone 2 and structures erected there must be designed accordingly.

4. HYDROLOGY

The results of an extensive survey of the site hydrology are given in the report IDO-22042-USGS (References 3 and 5).

The MTR-ETR-ATR site has been explored by a number of test wells which show the water table to lie approximately 470 feet below the surface (surface elevation 4930 feet, water table 4460 feet).

Since 1952, radioactive liquid effluent from MTR and ETR is discharged to a leaching bed located at the southwest corner of the site. The present (1964) disposal rate is 171,800,000 gallons of water per year containing approximately 2200 curies. When ATR is operating, this discharge will probably be doubled since ATR has an operating power level somewhat in excess of the sum of MTR and ETR.

At a depth of approximately 150 feet, a dense layer of basalt has sealed off the disposed water from the regional water table. Because the vertical migration of the water has been blocked, it has and probably will continue to spread horizontally. The perched water zone now covers an elliptical region 3000 feet wide by 6000 feet long with its axis running north-south. No traces of radioactive contamination have been detected in the water table under the MTR-ETR area. A number of test wells have been drilled to study the spread of the water from the leaching bed. Investigations of the site to date indicate that the regional water table is well protected by an impermeable layer of basalt at least 300 feet thick under the MTR-ETR-ATR site.

5. WEATHER MONITORING

A weather station has been installed east of the Test Reactor Area and is equipped with wind speed and direction instrumentation. These instruments supply signals to recorders located in the MTR-ETR-ATR control rooms and Central Facility Weather Station.

The information from this station is specific for Test Reactor Area permitting rapid evaluation of evacuation routes to be used in the event of a radioactive release in the area.

II-2. REFERENCES

1. 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).
2. R. L. Nace et al, Geography, Geology, and Water Resources of the National Reactor Testing Station, Idaho, Pt. 2, Geography and Geology, United States Department of the Interior, Geological Survey Water Resources Division, Ground Water Branch, Boise, Idaho, IDO-22033-USGS (1956).
3. P. H. Jones, Hydrology of Waste Disposal National Reactor Testing Station, Idaho, An Interim Report, U. S. Dept. of Interior Geological Survey, Idaho, IDO-220420-USGS (December 1961).
4. Official U. S. Population Figures (Comptons Pictured Encyclopedia) 1950 and 1960 Census.
5. Hydrology of Waste Disposal NRTS-Idaho, Annual Progress Report 1962, IDO-22044 USGS, April 1963.

III. REACTOR BUILDING

III-1 Building Arrangement

III-2 Containment

III-3 References

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III-1. BUILDING ARRANGEMENT

1. GENERAL DESCRIPTION

The reactor building is a complex of enclosed areas approximately 200 by 200 feet, extending a maximum of 60 feet above and 60 feet below grade. The average depth of the substructure is about 42 feet, and the average superstructure height is about 38 feet. Building elevations and pertinent cross sections are shown in Figures III-1-1, III-1-2, III-1-3, III-1-4, and III-1-5.

The first floor, shown in Figure III-1-6, is located at grade and is divided into the following areas: reactor area, storage canal area, utility area (including diesel-generator pit), general storage and laydown area, critical facility area, counting room, control rooms, and offices. The first and second mezzanine floors, shown in Figure III-1-7, are devoted to office and H & V equipment rooms. The first and second basements, shown in Figures III-1-8 and III-1-9, are utilized for experiment cubicles and equipment; electrical equipment, heat exchangers, demineralizer equipment, heating and ventilating equipment, and other associated items.

Access between basement levels and the first floor is by means of four enclosed, main stairwells, located approximately at the midpoint of each face of the building. In addition, a freight elevator is provided in the southwest corner of the reactor room, and a personnel elevator is provided in the southeast corner of the reactor room. Stairways are provided to the mezzanine levels from the first floor. Each major area is provided with access to adjacent areas within the building by means of at least one door. In addition, each area above grade, except for the counting room, and each stairwell leading from basement areas, except for stairwell Number 3, may be evacuated directly to the outside without passing through any other area. The counting room and stairwell Number 3 each have two alternate exits. Counting room evacuation may be either through the reactor area or critical facility area; stairwell Number 3 evacuation may be either through the storage canal area or utilities area.

1.1 Gastight Areas

The building is divided into two basic types of design, gastight and non-gastight. The gastight volumes have been selected to include all areas where an accidental contaminated release from the reactor or loop experiments must be contained or where sealed penetrations from such an area to an otherwise "clean" area would be difficult to maintain during normal operation. The gastight volume includes the following: Reactor Area, including freight and passenger elevators; Reactor Control Room; Instrument Repair Room; Data Readout Room; Office Number 110; First Basement; and Second Basement. These areas are illustrated in Figures III-1-10 through III-1-20.

1.2 Reactor Area

The reactor area, as shown in Figures III-1-4 and III-1-6, located in the south central portion of the building, is approximately 88 feet (N-S) by 100 feet (E-W). The area is bound on the east by offices; on the west by the critical facility area; and on the north by the storage canal area. Two sliding steel doors are

FIG. III-1-1 REACTOR BUILDING ELEVATIONS.

FIG. III-1-3 REACTOR BUILDING SECTION.

FIG. III-1-5 REACTOR BUILDING SECTIONS.

FIG. III-1-6 FIRST FLOOR PLAN.

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FIG. 111-1-7 FIRST AND SECOND MEZZANINE FLOORS.

FIG. III-1-9 SECOND BASEMENT FLOOR PLAN.

LEGEND OF SYMBOLS

- = GAS-TIGHT SURFACE
- = GAS-TIGHT FLOOR PENETRATION
- = GAS-TIGHT WALL PENETRATION
- = GAS-TIGHT DOOR
- = STEEL RE. (GAS-TIGHT)
- = CONCRETE (GAS-TIGHT)

FIG. III-1-12 GASTIGHT AREA SECTION.

FIG. III-1-13 GASTIGHT AREA SECTION.

FIG. III-1-14 GASTIGHT AREA SECTION.

FIG. III-1-15 GASTIGHT AREA SECTION.

FIG. III-1-16 GASTIGHT AREA -- FIRST BASEMENT PLAN.

FIG. III-1-17 GASTIGHT AREA -- SECOND BASEMENT PLAN.

FIG. III-1-18 GASTIGHT AREA -- MISCELLANEOUS PLANS.

located in the south wall to accommodate trucks. Access from the reactor area to the basement levels is provided by means of the freight and personnel elevators and stairwell Number 1, located along the south wall. The reactor is located about 66 feet from the south wall. The working canal is located north of and adjacent to the reactor vessel and connects directly to the storage canal. The nozzle access trench, reached by stairway from the reactor room, is concentrically arranged around the reactor with its outer walls extending three feet above the floor and joining the working canal parapet walls on the north side of the reactor. The reactor area is served by a 40-ton-capacity, overhead crane furnished with a 10-ton-capacity, auxiliary hoist. A 2-ton-capacity, bridge crane at a higher level with limited east-west coverage is also provided.

Floor area on three sides of the reactor serves for cask and other storage, truck access, maintenance, and handling operations. A large equipment hatch to service the basement levels is provided just south of the reactor.

1.3 Storage Canal Area

The storage canal area, shown in Figures III-1-5 and III-1-6, located north of the reactor area, is approximately 44 feet (N-S) by 200 feet (E-W) and extends 48 feet from floor to roof. A mezzanine level at the east end of the storage canal area supports heating and ventilating equipment. The canal length is approximately 156 feet (E-W). A movable bridge, spanning the canal and running on the parapet walls, is provided for access across canal and handling operations. Storage space for casks and other equipment is provided on the floor, north and south of the storage canal, and a truck aisle is provided north of the canal with rolling steel doors in the east and west building walls. The area is served by a 30-ton capacity bridge crane. An equipment hatch to serve the first basement is provided in the northwest corner of the area.

1.4 Utility Area

The utility area, shown in Figures III-1-5 and III-1-6, located in the north-east corner of the building, is approximately 68 feet (N-S) by 105 feet (E-W). Two 1500-kW diesel-generators are located on the north side in a 6-foot-deep pit that is 25 feet wide (N-S) by 88 feet (E-W). Heating and ventilating equipment is located in the southeast corner, and the process control room and other offices are located in the southwest corner. Fire walls separate the utility area from the storage canal area to the south and the storage and laydown area to the west. The exterior wall along the east side of the area is also a fire wall for protection from the main transformers, externally located east of this area. A five-ton capacity bridge crane is provided over the diesel generators for servicing and maintenance. Truck access to the area is provided by a rolling steel door in the north wall. An equipment hatch to serve the first basement level in the heat exchanger area is provided in the northwest corner.

1.5 General Storage and Laydown Area

The storage and laydown area shown in Figures III-1-3 and III-1-6, located west of the utility area, is approximately 68 feet wide (N-S) by 95 feet long (E-W). Space for general storage and a tool room is provided. Floor hatches are located over the heat exchangers and the primary coolant pumps, and a 10-ton monorail is located over the pump hatches. Removable roof hatches permit heat exchanger removal by a yard crane. Truck access to the area is by means of a rolling steel

door in the north wall. Fire walls separate the storage area from the storage canal area to the south and from the utility area to the east.

1.6 Critical Facility Area

The critical facility area, shown in Figures III-1-4 and III-1-6, located south of the storage canal area and west of the reactor area, is approximately 50 feet wide (E-W) by 66 feet long (N-S). The ATRC is located in an extension of the storage canal. A portable working bridge spans this portion of the canal and runs on top of the canal parapet walls. A movable control rod drive platform is provided directly over the critical facility during operation. Transfer of this platform between its operating position and a storage station is accomplished by a 15-ton capacity bridge crane serving this area. Critical facility instrument panels are located in the south end, and offices for the critical facility operators are on mezzanine level south of the critical facility area. Heating and ventilating equipment is located on the mezzanine level south of the ATRC. Each mezzanine is provided with an exterior exit and an enclosed stairway to the first floor.

The floor area around the critical facility is adequate for cask and other equipment storage. Truck access is by means of a rolling door in the west wall. Fire walls separate the critical facility area from the storage canal and reactor areas. The critical facility canal may be isolated from the storage canal by a vertical lift door and an insulated bulkhead.

1.7 Counting Room

The counting room, shown in Figures III-1-5 and III-1-6, with provision for 10 counting stations is in the southwest corner of the building and is approximately 26 feet wide (E-W) by 31 feet long (N-S). The room is shielded against background radiation by two-foot thick concrete walls, floor, and ceiling. It is also shielded from electrical fields by an electrostatic shielding envelope. A sample preparation laboratory is located east of the counting room.

1.8 Offices

The office area, Figures III-1-4, III-1-5, and III-1-6, located east of the reactor area, is approximately 50 feet wide (E-W) by 88 feet long (N-S) and is two stories high. The reactor control room, instrument repair and data readout rooms, and shift superintendent's office are all located on the first floor, north of the entry passage. The control room is adjacent to the reactor area, and windows are provided for viewing the reactor space. The health physicist's issue room, men's locker and shower room, and health physicist's shift office are located south of the main entry. A stairway off the entry hall leads to the mezzanine office area.

1.9 First Basement

The first basement, Figures III-1-5 and III-1-8, 19 feet below the first floor, is divided into three main areas. The northeast corner contains electrical switchgear, battery room, and other associated electrical equipment. Two exits from the area are provided by stairwell Numbers 2 and 3. The northwest corner contains the heat exchangers which stand in a pit 50 feet below the first floor, primary coolant pumps and motors, and hatches over the demineralizer vaults. A 10-ton capacity monorail is provided over the primary coolant pump motors, and a 6-ton-capacity monorail is provided over the demineralizer hatches. Two exits

from the area are provided by stairwell Numbers 3 and 4. The balance of the first basement floor, south of the canal walls, contains the reactor, the outer shim rod drive service area, safety rod drive service area, experiment cubicles, experiment sample rooms, experiment instrument and control panels, source room, and fuel storage room. Five exits from this area are provided by stairwell Numbers 1, 2, and 4, and the personnel and freight elevators.

1.10 Second Basement

The second basement, Figures III-1-5 and III-1-9, 38 feet below the first floor, is divided into three main areas. The northeast area contains the high-pressure demineralizer water heat exchangers and pumps. The northwest area contains the primary heat exchangers. Directly connected with this area is the pipe tunnel which runs south to the piping space and thence to the pipe corridor located on the east and west sides of the subpile room. The subpile room is located directly below the reactor. The balance of the second basement includes five experiment cubicles, experiment sample rooms, experiment control panels, auxiliary experiment equipment, the main exhaust fan room, and the demineralizer tank vaults. Each normally accessible area is provided with at least two exits. Sub-basement levels include the liquid-waste pump-motor room which is south of the reactor, 46 feet below the first floor; the hot and warm waste tank vaults which are 59 feet below the first floor and directly under the waste pump motor room; the neck-shim rod-drive room which is 58 feet below the first floor and directly under the subpile room; and the piping space which surrounds the neck-shim rod-drive room and is 58 feet below the first floor. Hatches are provided in the second basement and waste pump room floors for access to the hot and warm waste tanks.

2. SPECIAL FEATURES

2.1 Building Fire Protection Systems

The ATR building fire protection systems consist of automatic wet pipe sprinklers, fire hose stations, and various types of hand extinguishers. Automatic and manually operated alarm systems are also provided.

The building fire protection systems conform with requirements of the Uniform Building Code (UBC) of the International Conference of Building Officials, AEC-IDO Safety and Fire Protection Design Criteria Manual^[1], and regulations and requirements of the National Board of Fire Underwriters (NBFU) and National Fire Protection Association (NFPA).

Automatic wet pipe sprinkler systems are provided to protect the major populated and cubicle areas in the first and second basements, first floor, and mezzanine floors. These areas are also supplied with fire hose stations and various types of hand extinguishers. The sprinkler installations conform with requirements of NFPA Standard Number 13, and extinguisher and hose stations conform with requirements of NFPA Standard Numbers 10 and 14, respectively. Flow indicators and alarm circuits are provided for the sprinkler systems. Manual fire alarm boxes are provided throughout the building, and these,

together with the ADT alarm circuits, are connected with the existing MTR-ETR circuits and conform with requirements of NFPA Standard Number 72.

Automatic dry chemical systems with fire detector units protect the diesel generator installation and are supplemented by a portable dry chemical extinguisher and dry chemical hand extinguishers in the area. The automatic system conforms with requirements of NFPA Standard Number 17. ADT alarm circuits are also provided.

The following areas, while not provided with an automatic sprinkler system, are protected with fire detector units (temperature rise detectors) or fire detector heads (combustion products detectors), supplemented by fire hose stations and hand extinguishers:

- (1) Utility Area
- (2) General Storage and Laydown Area
- (3) Primary Coolant Heat Exchangers Area
- (4) Primary Coolant Pump Motor Area
- (5) Switchgear and Battery Areas
- (6) Fuel Storage Room.

The process and reactor control rooms and the counting room are protected by fire hose stations and hand extinguishers. Miscellaneous areas in the first and second basements which are not accessible during operation are protected by fire hose stations or hand extinguishers or both. These areas can be made accessible for fire-fighting purposes by scrambling the reactor.

2.2 Building Canals

2.21 Detailed Arrangements and Uses. A system of interconnected canals runs between the reactor area, storage canal area, and critical facility area. The arrangements are shown in Figures III-1-3 and III-1-6. The canals are functionally divided into four areas as follows: (a) reactor working canal, (b) storage canal, (c) the critical facility canal, and (d) the experimenters' working canal.

All canals are eight feet wide except for the critical facility canal which is 10 feet wide. A parapet wall extending three feet above the first floor forms part of the canal walls with the water level one foot below the top of the parapet. Depth of canal water is 20 feet except as noted in the following sections.

The reactor working canal, located adjacent to the reactor, contains a transfer device in a 45-foot-deep section. This device is used to transfer spent fuel, expended control elements, reactor internals, and capsule experiments between the reactor and the canal through the reactor drop tube. The loop cask transfer station is located over this 45-foot-deep section and is adjacent to the transfer device. The remainder of the reactor working canal is 22-foot-deep and contains a loop test assembly storage rack, a pressure tube storage rack, a handling tool rack, and one spent-fuel storage rack. The fuel storage racks,

with a capacity of 40 elements, will hold spent fuel elements during a cooling period after removal from the reactor before they are transferred to the shallower storage canal.

The storage canal, connected to the working canal, contains space for cask storage, irradiated hardware baskets, five fuel element storage racks for 200 elements (five cores), underwater saw and saw table, test assembly storage rack, and handling tool rack. The test assembly storage rack is located over a 39-foot-deep section at the east end of the canal to permit handling an entire loop experiment test assembly, preparatory to shipment out of the building. The west end of the canal includes provisions for future expansion. The total fuel element storage capacity of six reactor cores in the storage and working canals is sufficient to provide at least 102 days of cooling before transfer out of storage is required.

The critical facility canal contains a low-power, operating core mock-up of the ATR reactor core where nuclear measurements can be made to support ATR operations. The critical experiment is located in a 23-foot-deep section at the south end of the canal.

The experimenters' working canal contains a shielding-viewing window through which irradiated experiments can be observed prior to returning them to the reactor, shipping out of the ATR building, or moving to storage elsewhere in the canal. The center of the window is located five feet, four inches above the first basement floor. The window is constructed of high density glass slabs, laminated to a total thickness of about four feet. The wet face of the window is tempered glass plate sealed to the window frame and canal liner, and of sufficient thickness to withstand an average pressure of 16 feet of water. The dry face of the window is sealed, except when being used, by a steel plate which is also designed for the full water pressure. With the dry face of the window flush with the outer surface of the canal wall, the wet face is inset about three feet into the canal wall. This three-foot inset minimizes the possibility of window damage by handling operations in the canal. The total glass thickness, plus the water in the inset, gives shielding protection equivalent to the seven-foot-thick concrete canal wall. The window may be removed from the cold side in the first basement without draining the canal.

2.22 Construction and Operation. The canal is constructed of reinforced concrete walls and floor, with polyvinyl water stops provided in all construction joints. The canal is completely lined with seal-welded, stainless-steel sheets extending to the top of the parapet. A series of seepage drains are provided in the concrete wall behind the lining to locate leaks and to prevent any pressure buildup behind the liner plates.

The canal is filled with demineralized water and is purged by a continuous makeup and drain of 60 gpm. The canal water temperature, with six spent fuel cores in storage, is not expected to exceed 85°F because of the continuous purge flow, evaporation, etc; however, provisions have been made for insertion of cooling coils should this prove necessary. The critical facility canal is normally isolated by an insulated bulkhead, and its water is maintained at a higher temperature.

2.23 Drainage. The ATR canal system has been designed to prevent drainage by mechanical failure, accident, or operator error. The major features to control drainage are:

- (1) Conservative and proven construction methods
- (2) A system of outside (canal) drain valve, inside (canal) drain plate, and alarm (warm waste tank)
- (3) Administrative control of canal drainage
- (4) Time available to take corrective action.

Details of canal draining procedures and equipment are given below.

Watertight bulkheads are provided to permit sectional drainage or isolation of the following canal areas:

- (1) Critical Facility Canal
- (2) West End of Storage Canal
- (3) East End of Storage Canal
- (4) Experimenters' Canal
- (5) Working Canal.

The canal water level is normally maintained at a level one foot below the top of the parapet by five overflow drains, one located in each of the five main canal areas. Each area may be drained independently by insertion of the proper isolating bulkhead or bulkheads and using the drain in the bottom of the canal. Each drain is valved on the outside of the canal wall, and piping connects the drain with the warm waste tank which is located about 52 feet below the normal canal water level. In addition to the drain valves, flat circular plates are located in each drain line inside the canal. (These plates can be removed only by a canal handling tool and must be removed before canal drainage is possible.) Canal drainage is carried out under the administrative control of the operator.

The overflow drains act as skimmers to keep the water surface clean. In addition, a vacuum cleaning device is provided to clean accumulated debris from the canal. The skimmers and vacuum cleaning wastes are discharged to the warm waste system.

During normal operation, two bulkheads are in place: one at the north end of the critical facility canal, and one at the midpoint of the working canal. In addition, movable bulkheads are provided, which may be inserted to isolate any one portion of the canal from the balance of the canal. The following tabulation shows the time required to drain the canal to the top of a stored spent fuel element for normal bulkhead placement and for isolated section bulkhead placement:

<u>Location of Stored Fuel Element</u>	<u>Drop in Water Level</u>	<u>Time Required to Uncover Fuel [a]</u>	
		<u>Normal Operation</u>	<u>Isolated Section</u>
Working Canal	16.18 ft	128 min	14.6 min [b]
Storage Canal (East)	14.18 ft	112 min	27.8 min
Storage Canal (West)	14.18 ft	112 min	53.2 min
Critical Facility	14.18 ft	21.6 min	21.6 min

If the warm waste tank is empty when the drain flow starts, it will take less than three minutes for the tank's high-level alarm to signal in the process control room. In the worst case of the improbably isolated section conditions, there would be over 11 minutes after the alarm for corrective measures.

Based on the above factors, uncovering of fuel in the ATR canal system is not considered credible.

2.3 Building Design and Construction

2.31 Design Criteria. The reactor building design including the canals conforms with requirements of the Uniform Building Code (UBC) of the International Conference of Building Officials, 1961 Edition; AEC-ID Engineering Standards; American Concrete Institute (ACI) specifications; and American Institute of Steel Construction (AISC) specifications. Building design includes provision for seismic loads in accordance with provisions of the UBC for Zone Number 2 locations (generally, five percent of gravity loads); wind loads varying from 30 to 40 lb/ft², depending on height as specified in the AEC-ID standards for this area; actual equipment and dead loads; and liberal live load allowances to permit flexible operations in fuel and experiment handling, reactor operations, and storage and transfer of shielding materials, hatch covers, etc. Live load design values for the reactor building are shown in Figure III-1-21.

2.32 Substructure Construction. The building substructure is of heavily reinforced concrete construction, either resting directly on and keyed into the subsurface lava or supported by concrete piers which extend to the lava level. Shielding requirements are utilized wherever possible to meet part of the structural requirements of the building. While all load-carrying frames are of reinforced concrete, many partition walls and some shielding walls are of masonry (concrete block) construction. Where required, all masonry walls below grade are reinforced to resist seismic loads.

[a] Based on average drain flow of 1450 gpm

[b] With bulkhead at midpoint of working canal in raised position

FIG. III-1-21 REACTOR BUILDING LIVE-LOAD DIAGRAMS.

While most substructure areas are left unpainted, some (ie, the cubicle walls and floors, pipe corridors and tunnel, and subpile room) are protected with a phenolic type coating to simplify decontamination.

The basement areas are subdivided by fire walls where necessary to meet code requirements, and all stairwell and elevator enclosures are designed as fire walls.

2.33 Superstructure Construction. In general, the building above grade is of steel frame construction with exterior walls of insulated aluminum sandwich panels supported by steel girt and a reinforced masonry base wall, three feet, four inches high, around the building perimeter.

Roof construction consists of steel decking covered with a vapor barrier, insulation, and roofing. Generous slopes are provided over the entire area to internally located drains to handle the expected precipitation.

Gastight areas of the superstructure are of seal-welded steel plate, reinforced masonry, or concrete construction. Special provisions have been made to maintain the integrity of the confinement perimeter with respect to penetrations and doors.

Windows are provided only for the offices in the southeast corner of the building. Generally, swing doors are hollow metal, and truck access doors are of rolling steel shutter construction; however, the two truck doors in the south end of the reactor area are vertical lift doors of reinforced steel sheet construction, and the truck access door in the north wall of the reactor area is a horizontal sliding door of reinforced steel sheet construction. A large vertical-lift fire door of reinforced steel sheet construction is provided in the fire wall separating the critical facility and storage canal areas. Another large steel plate vertical-lift door is furnished over the working canal in the north wall of the reactor area. This door is furnished with a stainless steel bulkhead, mounted on its south face and so arranged that the bulkhead extends to the working canal floor when the door is closed.

III-2. CONTAINMENT

1. DESIGN CRITERIA FOR GASTIGHT AREAS

The ATR gastight areas were designed to partially contain fission products from a severe accident involving the reactor primary system and installed experiments. The maximum leak rate permitted from the gastight areas is ten percent by volume, per day. The actual leak rate will be determined by testing.

2. DETAILED DESCRIPTION OF GASTIGHT AREAS

The gastight areas are shown in Figure III-1-10 through III-1-20 and represent approximately 62 percent (1,640,000 cubic feet) of total building volume.

2.1 Substructure (1,122,900 cubic feet)

The substructure includes all first and second basement areas except stairwell Number 3 which has been excluded to permit first floor access to the process-utility control room on the first floor without entry into the gastight area.

2.2 Superstructure (517,000 cubic feet)

The superstructure includes the reactor area, reactor control room, instrument repair room, reactor data readout room, office Number 110, freight elevator, passenger elevator, and stairwell Numbers 1, 2, and 4.

Surfaces of the gastight volume identifying the portion exposed directly to the atmosphere are as follows:

<u>Material</u>	<u>Total Surface Area (ft²)</u>	<u>Portion of Surface Exposed Directly to Atmosphere (ft²)</u>
Welded Steel Plate	24,000	21,000
Masonry	11,000	2,300
Concrete	<u>106,000</u>	<u>0</u>
	141,000	23,300

2.3 Penetrations

Penetrations through the gastight perimeter surfaces have been minimized. Table III-2-1 gives the number of penetrations and sealing techniques employed in the design as illustrated in Figure III-1-19.

All penetrations are designed to be gastight and sealed either by caulking or gasketing. Fixed penetrations will maintain very high integrity and show practically no leakage. Most of the estimated building leakage will be through the doors and hatch covers. All swing doors are gasketed and provided with automatic closers. All sliding doors are provided with wedged or inflatable gaskets and are motorized. All hatches are gasketed.

TABLE III-2-1

NUMBER OF PENETRATIONS IN GASTIGHT AREA

(1) Substructure

- (1.1) Four electrical duct banks containing approximately 200 cables. Each duct is sealed with sealing-fitting for each cable.
- (1.2) Five pipes welded to embedded steel plates.
- (1.3) Twenty-two pipes through gasketed sleeves.
- (1.4) Two gasketed hollow metal doors with caulked frames.
- (1.5) One gasketed H&V exhaust duct sealed by a rubber-lined, quick-closing damper.

(2) First Floor

- (2.1) Fourteen gasketed hatch covers.
- (2.2) Five gasketed neoprene panels for tubing and cable seals.
- (2.3) Six steam pipes sealed by expansion joints.
- (2.4) Six gasketed H&V ducts sealed by rubber-lined, quick-closing butterfly dampers.
- (2.5) Eighty-one miscellaneous tubes, rods, pipes and conduits, sealed by fittings, or packed as required.
- (2.6) Two electrical bus ducts sealed by sealing fittings.
- (2.7) Four removable floor plugs.

(3) Superstructure

- (3.1) Eleven gasketed hollow metal doors with caulked frames.
- (3.2) Twelve caulked windows with fixed sash.
- (3.3) Three steel truck doors with inflatable gaskets.
- (3.4) One vertical-lift door in the north wall of reactor room over canal. This door is of solid steel plate with inflatable gaskets around the perimeter except for the bottom edge which extends two feet below canal water level. To maintain gas tightness in the event of a rupture of canal liner connection to the reactor drop tube which could drain the canal, a bulkhead, mounted on the south face of the door, extends to the bottom of the canal and operates as part of the door. The working canal forms the seal between the door and the bulkhead.

TABLE III-2-1 (continued)

- (3.5) Eleven gasketed H&V ducts and registers sealed by rubber-lined, quick-closing butterfly dampers.
 - (3.6) Four steam pipes sealed by expansion joints.
 - (3.7) Four hundred linear feet of gasketed and caulked intersection of steel plate liner with concrete and masonry surfaces.
-

2.4 Ventilating Systems

2.41 General. Ventilation for the principal working areas in the building is sectionalized to minimize the dispersal of activity from one area to another. In addition, the individual ventilation systems are independent of each other and generally may continue or discontinue operation without affecting other spaces. Fans and dampers controlling ventilation in specific areas will be secured by the Radiation Monitoring System (RMS) when necessary to confine a release of radioactivity (see Section VII-6.8).

The ventilation systems provide separate service to the principal working areas in accordance with accepted standards of the American Society of Heating, Refrigeration, and Air Conditioning Engineers.

All duct work which passes through fire walls or floor incorporates fire dampers at the point of penetration to ensure closure and to prevent carry-over of smoke in the event of fire.

All systems are diagrammatically illustrated in Figure III-2-1.

2.42 Criteria for Gastight Area Ventilation. In view of the importance of maintaining the integrity of the gastight area in the event of a release of activity, special criteria have been established for that portion of the ventilation system serving this area.

The gastight area is served by three supply systems as follows:

- (1) Reactor Area Supply
- (2) A branch from the utility area supply, that feeds the first and second basements
- (3) Control room air conditioning.

These systems and their associated dampers will be closed on a high activity signal from the RMS located in the reactor area, exhaust plenum, or stack.

The gastight area is served by a main exhaust system and roof ventilators. The roof ventilators are interlocked to the RMS in a manner identical with the supply systems. The main exhaust system serves as a collection plenum for the discharge of several specific area supplementary fans. Each of the supplementary exhaust fans is interlocked to stop when the main exhaust fans are

FIG. III-2-1 H & V SCHEMATIC DIAGRAM.

shut off (only after a suitable time delay, to allow for automatic start-up of the standby main exhaust fan if the initiating signal is other than RMS). The main exhaust fan will be shut off by either of two independent RMS signals, so that the high activity is not discharged out of the ventilation stack.

2.43 Reactor Operating Area and Basements. Ventilation air is supplied directly to the reactor operating area and the first and second basements by means of a duct system which then distributes the air through double deflection grilles.

The equipment for this system is located in the heating and ventilation room on the second mezzanine level in the southwest corner of the building. It consists of a filtering system; preheat and reheat steam coils; air washer; and supply fan, HVS-1. The fan provides the reactor operating area with six air changes per hour and provides the basement levels with four air changes per hour.

A fan, HVE-2, located on the first basement level is used to transfer air from the clean first basement area to the second basement to provide four air changes per hour for both levels. This fan is interlocked to stop when the main exhaust fans stop.

In the summer, 100 percent outside air is furnished. A roof ventilator, HVR-1, is utilized to exhaust approximately 28.5 percent of the supply to the reactor operating area. In the winter, this quantity is recirculated.

The reactor room exhaust fans, HVE-1A and 1-B, one a standby, discharge an additional 43 percent to the main exhaust system. In the event of the failure of the operating fan, the standby is automatically started. The reactor room exhaust system duct work is furnished with shutoff dampers and a bypass duct connection to discharge to the main exhaust system via either the primary heat exchanger area (normal flow) or directly to the main exhaust plenum in the event of spillage in the reactor operating area and/or when maintenance in the heat exchanger area requires air bypass to the plenum.

The balance of approximately 28.5 percent is utilized for reactor shield cooling by exhaust fan, HVE-7.

The reactor room exhaust fan and reactor shield cooling exhaust fan both operate continuously and are interlocked to stop only when the main exhaust system is stopped.

Since the reactor operating area is part of the gastight enclosure, butterfly-type dampers are installed in the supply and return air duct work and in the inlet to the roof ventilators to ensure integrity of enclosure on shutdown and to preclude leakage to atmosphere. The leak tight butterfly dampers, which can close within three seconds, are automatically controlled by the operation of the RMS. The supply fans are interlocked to the RMS and, as a backup, to the operation of the main exhaust fans. The RMS includes continuous air monitors and beta-gamma monitors located in the reactor operating area and basement levels and gamma monitors located at the stack and exhaust plenum. These instruments control the operation of the reactor area supply and exhaust systems to balance potential internal pressure build-ups and yet contain all high radioactivity.

The reactor room exhaust fans, HVE-1A and 1B, can be utilized to purge the reactor operating area in the event of local spills where radioactive material is released. The purging of the area can be accomplished, with no other system operating, by opening either or both of the large rolling doors and drawing air from outside across the reactor area. This air can be discharged directly to the main exhaust plenum by manually controlling the bypass damper.

The first and second basement areas are provided with ventilation from the reactor area supply system, a branch of the utility area supply system, and the reactor area exhaust system. The supply air is furnished to clean areas, and the basement exhaust systems use supplementary exhaust fans which ensure the direction of air flow from clean areas to possibly contaminated areas.

The loop experiment cubicles are maintained at a negative pressure through the use of cubicle exhaust fans, HVE-5 and 6, which draw air from the clean basement areas through the adjacent sample areas and into the cubicles through offset wall openings. Fan HVE-6 serves all the cubicles on the east side of the reactor, and fan HVE-5 serves all the cubicles on the west side of the reactor. Each cubicle exhaust duct is connected to a header through a manually operated butterfly-type damper. This damper can function to adjust the negative pressure maintained in each cubicle. In addition, this damper can be used to isolate individual cubicles from the exhaust system. The cubicle exhaust fans discharge air from the header to the plenum of the main exhaust system. The cubicle exhaust fans stop when the main supply fan is shut off.

In addition to the exhaust ventilation arrangement for the cubicles, each cubicle is provided with an air handling unit with water cooling coils to dissipate 35 kW of heat from internal pipe and equipment.

The rod drive service areas and the piping corridors which surround the reactor vessel are provided with ventilation through a branch duct from the basement supply system. The quantity of air supplied is based on removing the pipe heat losses and provides six air changes per hour. The air is exhausted from these areas through a duct system connected to a supplement fan, HVE-16, which discharges into the main exhaust system. This fan stops when the main supply fan is shut off.

The subpile room, rod access room, and motor and tank areas of the warm waste and hot waste vaults are ventilated by exhaust fan HVE-8. The fan draws air from the clean area of the second basement level and discharges into the main exhaust system. The arrangement of duct work ensures a negative pressure in the rooms and vaults to prevent backflow of contaminants. This fan stops when the main exhaust fans are shut off.

The primary coolant pump motor area is ventilated, in part, by transfer fans, HVE-3A and B; one is a standby. Additional ventilation for this area is provided through the branch from the utility system. The total quantity of air circulated in the primary pump motor area is exhausted to the primary heat exchanger area. The standby fan is automatically started in the event of failure of the operating fan.

The degassing tank is maintained at a negative pressure through the use of exhaust fan, HVE-18, which draws air from the primary heat-exchanger area

through the tank. The fan discharges air into the plenum of the main exhaust system. A beta-gamma monitor is used in the degassing system to sound an alarm on high activity level. This monitor is not interlocked with the exhaust system. The degassing tank exhaust fan stops when the main exhaust fans are shut off.

The primary heat exchanger area is ventilated by exhaust fan, HVE-4, which draws air in from the basement and discharges into the main exhaust system. This fan prevents backflow of contaminants from the primary heat-exchanger area to the basement and also ensures proper air flow direction in the primary coolant pump motor area. This exhaust fan stops when the main exhaust fans are shut off.

2.44 Main Exhaust System. The main exhaust system fans are located in the equipment room on the second basement level. The fans, HVE-17A and B, one a standby, ensure continuous operation. In the event of failure of the operating fan, the standby is automatically started. Each fan is sized to handle all input air at a pressure that will ensure some flow in each circuit even if supplementary fans are inoperative. All of the exhaust air from the various supplementary exhaust fans is collected in an exhaust plenum which is directly connected to the main exhaust fan intake. The fan outlets are combined into a common discharge duct which is then carried below grade to the ventilation stack. The duct below grade is constructed of concrete pipe to minimize corrosion and maintenance problems. The discharge duct is brought above-grade, approximately 50 feet from the stack centerline, to permit the inclusion of a filter bank or scrubber in the event future experimental programs indicate the need. A butterfly-type damper is included in the duct at the exhaust stack to enable tight shutoff of the exhaust line. These fans operate continuously until shutdown by a high-level signal from either the stack monitor or the exhaust plenum monitor. When the main exhaust fan is stopped, all the interlocked supplementary exhaust fans are automatically stopped. The butterfly damper at the stack is independently interlocked to RMS to close from either the stack monitor or, after a time delay, the gas monitor in the exhaust plenum. Separate interlocks are provided so that the damper does not close on loss of the main exhaust fans unless the RMS readings are high.

The ventilation stack is a 250-foot-high, tapered, unlined, reinforced concrete structure which handles 64,800 cfm of exhaust air at a maximum temperature of 105°F. The stack is approximately 300 feet northwest of the ATR building. The inside diameter of the stack is five feet at the top, resulting in a discharge velocity of 3300 feet per minute which permits good mixing with atmospheric air and excellent wind carry-off. The stack has a concrete shell thickness of eight inches minimum and is designed to resist stresses due to the weight of the shaft, seismic forces, wind forces, and temperature differentials.

2.45 Critical Facility Area. Air to the critical facility area and other rooms on the ground floor and first mezzanine level is provided by a separate system.

The equipment, located in the heating and ventilating room on the first mezzanine level in the southwest corner of the building, consists of a filtering system, steam heating coils, and supply fan, HVS-2. The fan provides the facilities area with four air changes per hour, the offices and sample preparation

laboratory with 10 air changes per hour, and the toilets with 15 air changes per hour.

In the summer, 100 percent outside air is furnished. A roof ventilator, HVR-2, is utilized to exhaust the air supplied to the facilities area and the offices. In the winter, the roof ventilator is closed and the air is recirculated. The toilets are provided with a continual year-round exhaust fan system, HVE-11, which discharges to the atmosphere. A separate fan, HVE-15, exhausts air through an absolute filter from the hoods in the sample preparation laboratory.

This system is outside the gastight enclosure. Consequently, no special dampers are included. The roof ventilator and the exhaust fans are controlled by operation of the supply fan. When the supply fan is off, the ventilator and exhaust fans are also off. An interlock to the critical facility is actuated by an RMS continuous air monitor to shutdown the system.

2.46 Counting Room. This room is provided with air conditioning to maintain 75°F dry bulb temperature.

The air conditioner, HVA-1, is located in the heating and ventilating equipment room on the first mezzanine level in the southwest corner of the building.

The air is supplied through duct work to the space above the acoustical tile ceiling for diffusion to the room through the tile perforations. A fixed quantity of 20 percent outside air is mixed with 80 percent recirculated return air. A quantity of air equal to 20 percent of the total is supplied as makeup to the sample preparation laboratory.

This system is outside the gastight enclosure, and therefore, no special dampers or controls are included.

2.47 Storage Canal Area and General Storage Area. The supply air equipment is located on the platform at the east end of the storage canal area and consists of a filtering system; steam heating coils; and supply fan, HVS-3. The fan provides these areas with four air changes per hour.

In the summer, 100 percent outside air is supplied and roof ventilators, HVR-3 and 4, are utilized to exhaust the air. In the winter, the roof ventilators are closed and approximately 70 percent of the air is recirculated. The quantity of air supplied to the general storage area, approximately 30 percent of the system total, is exhausted the year round through a gravity-type roof ventilator, HVG-1, and exhaust fan, HVE-19. This fan also exhausts air from hoods over nitric acid storage carboys. The exhaust fan is manually controlled. This supply system is outside the gastight enclosure. Consequently, its operation is controlled by conditions within the space served and there is an interlock to the RMS. Either of the two CAM's, CAM 8 or CAM 9, located in this area can actuate the interlocks. In the event of local contamination spills, the storage canal area can be purged by opening the large rolling doors at either end and operating the roof ventilators.

2.48 Reactor Control Room, Data Readout Room, and Instrument Repair Room. A separate air-conditioning system maintains 75°F dry bulb temperature during summer and winter operation in these areas.

The air-conditioning unit, HVA-2, is located on the platform at the east end of the storage canal area.

The air is supplied to the rooms through a duct system and discharged through double deflection grilles. A fixed quantity of 20 percent outside air is mixed with 80 percent recirculated air. A quantity of air equal to 20 percent of the total supplied is exhausted through a relief duct to the toilet and locker room.

Since this system serves spaces which are part of the gastight enclosure, butterfly-type dampers are installed in the supply and return air duct work and in the relief duct, to ensure integrity of gastight volume. The butterfly-type dampers are controlled by the reactor area RMS system or, as a backup, upon shutdown of the main exhaust fan.

2.49 Offices, Health Physics Facilities, Toilet, and Locker Room. The supply air equipment for these areas is located on a platform at the east end of the storage canal area and consists of a filtering system; steam heating coils; and supply fan, HVS-4. The fan provides the offices and health physics facilities with 10 air changes per hour and the toilets and locker room with 15 changes of air per hour.

In the summer, 100 percent outside air is furnished. A roof ventilator, HVR-8, is utilized to exhaust the air from the toilets and locker room all year round. A supplementary fan, HVE-14, is used to exhaust air continually through a hood in the health physics office. The duct work is arranged so that the system return air, approximately 75 percent, may be bypassed to a supplementary exhaust fan, HVE-12, in the summer or recirculated in the winter. This system serves spaces which are not a part of the gastight enclosure. Consequently, no special shutoff provisions are included.

2.410 Utility Area. A ventilation system is provided to supply air to the utility area and adjacent spaces on the first floor, the switchgear and primary pump motor areas in the first basement, and the auxiliary and water equipment areas in the second basement.

The equipment is located in the southeast corner of the utility area and consists of a filtering system; steam heating coils; air washer; and supply fan, HVS-5. Sufficient supply air is provided to remove heat rejected by the diesel generators in the utility area on the first floor and heat generated by the switchgear in the first basement, to provide the second basement with four air changes per hour and to provide occupied utility spaces on the first floor with 10 air changes per hour. In addition, this system provides part of the air required to cool the primary coolant pump motor area.

In the summer, 100 percent outside air is passed through the air washer to decrease the temperature. Roof ventilators, HVR-5, 6, and 7 are utilized to exhaust the air from the first floor utility area. The air supplied to the primary circulating pump motor area is exhausted to the heat exchanger area and then

to the main exhaust system. A fan, HVE-9, is utilized to provide a continuous exhaust from the battery room. Air from the second basement is collected and returned in a duct system with a damper arrangement for recirculation in the winter or bypass to a supplementary exhaust fan, HVE-10, for discharge to the atmosphere in the summer.

The air supplied to the utility spaces on the ground floor is continuously exhausted by a supplementary exhaust fan, HVE-13. An absolute filter is included in the duct connection from the hood in the waste laboratory to this exhaust system.

The supply system serves the utility area which is outside the gastight enclosure and the basement areas which are within the gastight boundaries. In order to ensure the maintenance of gas tightness, the supply and return air duct work connecting this system to the basement areas and the battery room exhaust duct are provided with butterfly-type dampers. These dampers are controlled by the reactor area RMS or, as a backup, by the stack RMS.

2.411 Process Control Room. Air conditioning maintains 75°F dry bulb temperature during summer and winter operation.

The air-conditioning unit, HVA-3, is located in the utility area outside the east wall of the control room.

The air is supplied to the room through a duct and double deflection grilles. A fixed quantity of 20 percent outside air is mixed with 80 percent recirculation air. Twenty percent of the total air supplied is relieved to the adjacent stair hall. This system is outside the gastight enclosure; consequently, no special dampers or controls are included.

2.412 Power Supply for Ventilation System. All the supply and exhaust ventilation systems, except as noted, are furnished with commercial power for operation. The following fans and equipment are furnished with diesel power:

(1) Reactor Area Supply System	HVS-1
(2) Utility Area Supply System	HVS-5
(3) Washer Circulating Pump Associated with HVS-5	
(4) Utility Area Roof Ventilators	HVR-5, 6, and 7
(5) Reactor Room Exhaust Fan	HVE-1A or 1B
(6) Transfer Fan	HVE-3A or 3B
(7) Main Exhaust Fan	HVE-17A or 17B

The electrical service for systems, HVE-1A and B, HVE-3A and B, HVE-17A and B, includes the automatic switchover to the diesel driven unit in the event of failure of commercial driven unit and the reverse. All gastight area

shutoff dampers and other interlocks are supplied from the diesel-commercial automatic throw-over bus.

2.413 Ventilation Control. Each ventilation system damper related to the gastight area is controlled by a pneumatic piston that is supplied air through a three-way solenoid valve. Air is supplied to the piston to open the damper. Loss of air closes the damper by spring action. Loss of power causes the solenoid valve to vent air from the piston thereby closing the damper. An emergency panel has been provided in the diesel-generator room which includes starting buttons for the items noted in Section 2.412. This panel permits overriding all RMS interlocks.

2.5 Spray System

A spray system, arranged to wash down the walls and air inside the reactor area, is provided to reduce the concentration of fission products in the air contained in the gastight reactor area if a severe nuclear accident should occur. The fission product leakage from the gastight area to the outside atmosphere is therefore reduced. The reduction is especially effective for the radioactive iodines.

A three-inch line from the ATR fire-water loop provides water to the spray system in the reactor area. This line enters the reactor building in the first basement. Drainage is to the liquid waste tank.

The system is put into operation or shut down by either one of the two sets of remote push buttons actuating solenoid valve SOV-1. One set of push buttons is located on the reactor control panel, and the other set is located outside the crash gate at the southeast corner of the MTR-ETR area. The latter station is padlocked with the key retained in ID Security Headquarters.

3. ANALYSIS OF ACCIDENTS CAUSING A PRESSURE RISE IN THE GASTIGHT AREAS

3.1 Accidents Causing a Pressure Rise

The building internal pressure rise as a result of a release of coolant from a pressurized water loop or from the reactor primary system has been analyzed for the following accidents. The calculations were based on exhaust and supply of 58,400 cfm. Actual values of 64,800 cfm do not affect analysis conclusions.

- (1) Release of pressurized water loop contents to the reactor room
- (2) Release of pressurized water loop contents into the basements
- (3) Release of water from the reactor primary system into the reactor room
- (4) Release of water from the primary system into the basements.

Figure III-2-2 is a block diagram of the volumetric contents of the various areas comprising the ATR gastight volume and indicates the interconnecting

FIG. III-2-2 BLOCK DIAGRAM VOLUMES OF VARIOUS GASTIGHT AREAS.

flow paths. Because of the flow paths, a volume of gas released in any one space will be distributed to the entire gastight volume.

The ventilation exhaust system includes 18,600 cubic feet of storage volume of clean air which permits operation of the fans to exhaust a volume from the gastight areas of at least 18,600 cubic feet before the damper at the stack is closed by high radioactivity. If it is assumed the fluid releases to the reactor room are radioactive, the exhaust system will operate for not less than one minute before the radioactivity reaches the RMS at the stack damper at which time the damper will be closed. For releases to the basement, the exhaust system will operate not less than 18 seconds before the stack damper is closed by radioactivity. These time periods and the volume of stored clean air are used in the following analyses.

In each of the above cases, it is assumed that the supply system is shut off within three seconds as a result of high radioactivity signal in the reactor area. If the supply system is not shut off because radioactivity is not released coincident with the fluid, then the exhaust system is not as effective in relieving pressure rise. The volumetric relationships between the supply and exhaust systems will inherently adjust to relieve the pressure rise. The upper pressure limit is the shutoff head of the supply fans. The rate of fluid release, in the absence of radioactivity, in no case exceeds the rate at which the exhaust system can discharge air when the supply fans are at shutoff head. Accordingly, only the cases where radioactivity is released with the fluid and where the supply fans are shut off are discussed.

During normal operation, the reactor room will be at a temperature of 80°F dry bulb and 59°F wet bulb (26 percent relative humidity). This is approximately a 10-degree rise over the maximum temperature of air supplied from the air washer incorporated in the supply system. The rise primarily results from the sensible heat increase from solar transmission, equipment, and lighting loads. The basements will be stabilized at a temperature of 90°F, dry bulb and 63°F, wet bulb (25 percent relative humidity). This rise results from the large heat loads from equipment, motors, and piping in these areas.

3.11 Release of Pressurized Water Loop Contents to the Reactor Room. A pressurized water loop has a normal water volume of 225 gallons at 600°F and 2200 psig during operation. The system includes a primary pressurizer which contains 6.4 cubic feet of steam at 650°F and 2200 psig to maintain system operating pressure.

The construction of the portion of the loop extending through the top head of the reactor is such that the maximum size of a possible rupture is limited by the conax fitting. This limits the size of the opening to a clearance of approximately 0.2 square inch. The failure of the conax seal has been used as the basis of the analysis.

It is assumed that the conax seal at the top of the loop fails instantaneously to expose a clear discharge opening of 0.5-inch diameter to the reactor room. The contents of the loop will be discharged in 58.5 seconds.

The 225 gallons of loop water will flash into wet steam with a volume of 36,000 cubic feet. This is forced to mix with the confined volume of air in the gastight reactor room tending to raise the internal pressure in the building. In addition, while the loop contents are being discharged, the ventilation supply system is shut off by the RMS after three seconds, adding 3100 cubic feet to the gastight volume. However, the exhaust system will continue to operate until the stack monitor shuts it down. With a flow delay time of 60 seconds between the reactor room and the stack, the reactor room exhaust fans will, in this period, exhaust 20,000 cubic feet to the main exhaust plenum. The remaining 19,100 cubic feet will be discharged into the basement area through the 19 square feet of openings in the reactor room floor and will be removed by the ventilation system. The main 64,800-cfm exhaust system has the capacity to exhaust these volume additions to the stack before the RMS closes the stack damper. Accordingly, there will be no pressure rise in the building due to this accident.

3.12 Release of Pressurized Water Loop Contents into the Basement. A release of loop contents into the cubicles, piping corridor, or subpile room through a two-inch-diameter pipe rupture will produce a volume of vapor less than that released from the top plug rupture because the volume of air in the spaces where loop piping is located is small. These small volumes can accept only a limited amount of water vapor to reach saturation. Hence, it is assumed that the wet steam released displaces a volume of air equal to the volume of wet steam produced. This volume is 18,000 cubic feet. This release will occur in 12 seconds. No credit is taken for condensation on the cooler basement surfaces. During the 12-second period, the exhaust system is capable of removing 11,700 cubic feet based upon its discharge rate of 64,800 cfm, leaving 6300 cubic feet excess in the building. Additionally, the supply fans add 3100 cubic feet before the radiation monitor at the exhaust plenum shuts off this system leaving a total excess volume of 9400 cubic feet -- 6900 cubic feet of this total can be discharged by the exhaust system until the stored clean air is completely vented and the stack damper closes. Thus, the excess vapor remaining in the building after complete shutoff is 2500 cubic feet which will create an internal pressure of 0.019 psi. This has a negligible effect on both the building integrity and the average building leak rate in 24 hours, since this excess volume is rapidly dissipated.

3.13 Release of Water from the Reactor Primary System into the Reactor Room. The reactor vessel has a total water content of 2870 cubic feet of which 1850 cubic feet is above the core. It is assumed that the released reactor vessel water is equal to the expanded air volume in the surge tank, 1350 cubic feet. One-half of the released water will be ejected into the reactor room; the rest will be deflected by the arrangement of the vessel head and surrounding concrete structure and will drain into the shield cooling annulus.

Evaporation of the expelled water will heat the air and increase its moisture content. The quantity of water evaporated was calculated assuming that the water was sprayed into the air, then covered the entire floor area of the reactor room, and drained to the basement.

During the initial expulsion, the water is sprayed into the air and 38.4 pounds evaporates. The remainder of the water spreads out on the reactor room floor and requires 6.67 minutes to drain to the basements. During the 6.67 minutes,

an additional 550 pounds are evaporated. The water which drains to the basement eventually is trapped in the subpile room, the elevator shaft, and other pits. While draining to the basement and dropping through the air, an additional 38.4 pounds are evaporated.

While the water is in the reactor room, 14,670 cubic feet of vapor is released; and thereafter, 1930 cubic feet more is released in the basement. For analysis, it is assumed that the total volume is released in 6.67 minutes.

During the course of the accident, the RMS shuts off the ventilation supply system after three seconds which adds 3100 cubic feet of air to the building. The exhaust system continues to operate for 60 seconds before radioactivity closes the damper at the stack.

The total volume released is 19,700 cubic feet, the sum of 14,670, 1930 and 3100. The 19,700 cubic feet is accumulated in 6.67 minutes, assumed at an average rate of 2500 cfm. The exhaust system normally can exhaust 64,800 cubic feet per minute; however, if the supply system is shut off, the exhaust fans will approach their shutoff head; and it is estimated that only 33,900 cubic feet will be discharged in the minute that the exhaust fan continues to operate. Since this is greater than the vapor formed, the reactor room pressure will be negative. Accordingly, it is expected that some in-leakage will occur during a portion of the remaining 5.67 minutes that vapor is being generated at an average rate of less than 2500 cfm. At some point during this time period, the reactor room pressure becomes positive and out-leakage will commence to relieve the vapor. If it is assumed that out-leakage commences after 50 percent of the 5.67-minute period, the total out-leakage is limited to 7000 cubic feet. This is equivalent to less than 0.5 percent of the building volume per day. Furthermore, the peak pressure produced cannot be greater than 0.0525 psi which is equivalent to an instantaneous net addition of 7000 cubic feet to the building. This is well within the design capability of the building.

3.14 Release of Water from the Primary System into the Basement. The primary system piping is located in the piping corridor, pipe tunnel, and the heat exchanger room. A large rupture of the primary piping will drain the contents of the primary system into these spaces. The average temperature of the water is 150°F, and, for analysis, it has been assumed to collect in the heat exchanger pit.

The hot water in the pit will heat the air in the heat exchanger room and eventually saturate it. The heated and saturated air will expand and displace air from the tunnel. The volume of the heat exchanger room is 75,000 cubic feet compared to total tunnel and room volume of 158,000 cubic feet.

The evaporation of the water will produce 16,000 cubic feet of vapor in the first 30 minutes, 7000 cubic feet in the next 30 minutes, and 22,000 cubic feet more in the following 23 hours.

The vapor generated during the last 23 hours will leak to the atmosphere at essentially a uniform rate of 1000 cubic feet per hour. No significant pressure rise is associated with this vapor formation.

The exhaust fan and damper will be shut off at the time that the clean air volume of 18,600 cubic feet in the plenum and tunnel have been exhausted. At this time, the gastight area will be at a negative pressure. This will be relieved gradually by in-leakage and vapor production. If it is assumed that atmospheric pressure is reached in 15 minutes, the ensuing out-leakage over the next 15 minutes is limited to 7500 cubic feet. This is based upon a vapor production rate of 500 cfm, which is the average rate equivalent to 16,000 cubic feet during the first 30 minutes. The leakage during the following 30-minute period is limited to 7000 cubic feet and during the following 23 hours is limited to 22,000 cubic feet, for a total 24-hour leakage of 36,500 cubic feet. This is a maximum leakage rate of approximately 2.25 percent of the building volume per day.

The vapor produced during the first 30 minutes will govern the peak building pressure because it is produced at the greatest rate. The pressure associated with an instantaneous net addition of 7500 cubic feet is less than 0.056 psi. This is well within the design capability of the building.

3.2 Summary of Pressure Rise Analysis

The greatest pressure rise as a result of any of the accidents postulated is less than 0.056 psi. The pressure rises are relieved by the ventilation system and/or out-leakage from the building without impairing either the integrity of the gastight volume or exceeding acceptable leak rates.

4. ANALYSIS OF LEAKAGE FROM GASTIGHT VOLUME

Air leakage from the gastight building is a function of the pressure differential between the confined air volume and the surrounding atmosphere. If the pressure differential is maintained at zero, the only theoretical leakage would be by diffusion which is negligible. As the pressure differential increases, the building leakage also increases. Factors which affect this pressure differential are the atmospheric changes in temperature and pressure, the wind acting on the building, and the reactor accident. While a building leak test will be performed to determine the actual leak-rate, empirical methods are available for predicting the approximate leak-rate of a building [2]. These methods were used to calculate the theoretical leak-rates described in the following sections.

4.1 Factors Influencing Leakage

4.1.1 Effect of Changes in Atmospheric Temperature and Pressure. Building leakage, due to atmospheric temperature and pressure changes, is limited to the volume of air release necessary to equalize the confined building pressure with the atmospheric pressure. The greatest possible differential pressure and consequent maximum leakage will occur on a day that the maximum temperature and barometric changes are concurrent in their effect on the confined building volume. Data, gathered at the U. S. Weather Bureau Office at NRTS Central Facilities [3], indicates the maximum 24-hour temperature range to be 58°F. Calculations show a corresponding confined air temperature range of 24°F in the building. Due to this temperature change, a release of 75,000 cubic feet in 24 hours or 4.6 percent of the confined volume is required to maintain the differential pressure at zero.

The maximum barometric pressure change recorded by the weather bureau at NRTS over a 24-hour period is 0.5 inch Hg. Due to this pressure change, a release of 34,000 cubic feet in 24 hours or 2.1 percent of the confined volume is required to maintain the differential pressure at zero.

The combined effect of temperature and pressure changes acting concurrently requires a release of 109,000 cubic feet or 6.7 percent of the confined volume to maintain a differential pressure of zero.

4.12 Effect of Wind. The maximum seasonal average wind velocity, from recorded data[3], is 11.7 miles per hour. In establishing this maximum velocity, the lapse and inversion wind roses were averaged separately for each season. This is equivalent to a pressure differential of 0.087 inch H₂O on any wall of the building. The south wall of the reactor area is the only wall of the gastight volume which is fully exposed to wind action. It also contains more penetrations subject to leakage than any other wall. Therefore, the 11.7-mile-per-hour wind acting on this wall continuously over a 24-hour period will yield the greatest leak rate. This condition requires a release of 32,000 cubic feet in 24 hours or 2 percent of the confined volume in order to maintain a differential pressure of zero.

4.13 Effect of Reactor Accident. The most critical accident for leakage from the confined building volume is the accidental rupture of the reactor primary system as previously described. A release of 36,500 cubic feet in 24 hours or 2.2 percent of the confined volume is required to maintain the pressure differential at zero.

4.2 Analysis of Combined Building Leak Rate

Combination of the temperature, barometer, wind, and reactor accident factors previously described indicates a possible building leakage of 176,000 cubic feet or 10.75 percent for the first day. This combination of the accident concurrent with the maximum recorded temperature range coincident with both the maximum recorded barometric change and the maximum average wind velocity continuous over a 24-hour period is conservative. Accordingly, 10.75 percent is considered an upper limit leak rate for the first day. The corresponding upper limit for following days would be 8.52 percent. Furthermore, the 11.7 mile-per-hour wind during a nuclear accident, while giving a high leak rate, results in a high dispersion factor which reduces the radiation hazard to nearby populated areas.

The following are selected as a more probable combination of adverse meteorological conditions over the time scale of the accident:

Average daily temperature range, 38°F,[3]
(Corresponding confined air temperature range, 16°F)

Daily barometric change, 1/4 inch Hg

Continuous wind velocity, 2-3/4 miles per hour.

The maximum building leakage calculated under these conditions is 6.68 percent on the first day and 4.5 percent per day thereafter.

4.3 Effect of Pressurizing Factors on Building Frame

All analyses of fluid releases to the building are based on shutdown of the H&V supply fans by the RMS system. Resulting pressure buildup in the building, considering the time factor and computed leakage, is negligible insofar as building stress is concerned.

If, however, the fluid is released without radioactivity, the supply system will not shut down. In this case, only the magnitude of the pressure buildup and its affect on the building frame require consideration. This pressure buildup is limited to the shutoff head of the supply system, six inches of water. The stress in the building frame, due to the differential pressure, will not exceed 80 percent of its yield strength. This approximates the allowable stresses permitted for other transient loading conditions such as wind or earthquake, and it will be relieved rapidly by the inherent ventilation system flow characteristics. The meteorological effects over this time period are negligible. The meteorological factors, independently considered, are not significant in the building stress analysis.

5. BUILDING LEAK RATE TEST

A test will be made to determine the leak rate from the entire gastight area. The test result, expressed as the fraction of the total gastight volume leaking in a 24-hour period, will be used to verify the gastight design.

The test consists of adding helium to the gastight area in sufficient concentration to be accurately measured. The helium is dispersed throughout the gastight volume using fans and natural circulation. After obtaining reasonable helium dispersion throughout the volume, the helium concentration is measured as a function of time. Atmospheric conditions inside and outside the building are measured during the test. Entries in and out of the gastight area are noted during the test.

5.1 Test Preparation

Before starting the test, an inspection of all penetrations, doors, windows, and hatches, including seals and gasketing, will be made. The H&V system will be checked for performance and modified to help disperse the helium within the gastight volume.

5.2 Equipment

The following equipment and instruments will be used during the test:

- (1) Two Consolidated Electrodynamic leak detectors to determine relative helium concentration prior to starting test
- (2) One mass spectrometer located in ICPP and evacuated sample bombs
- (3) Two each, thermometers and mercury barometers to be located inside and outside the building
- (4) Wind velocity and direction indicator located outside the building
- (5) Twenty-five 180-cubic-foot helium cylinders.

5.3 Test Procedure

The test will be run when weather conditions are predicted to be stable. Helium will be dispersed to the gastight volume by bleeding cylinders into the supply fan (HVS-1) plenum chamber. All outside supply and return air dampers will be shut. Several additional fans will be used to disperse the helium uniformly throughout the gastight area.

Preliminary checks will be made to determine leakage areas that may have been missed during the inspection. These areas will be repaired or modified before starting the test. Following helium dispersion, as indicated by reasonably uniform detector readings throughout the volume, the test will be started. Periodic samples of the gastight volume at specified locations will be taken and analyzed in the mass spectrometer. Simultaneous readings of temperature and pressure inside and outside the building will be taken together with readings of outside wind velocity and direction. Personnel entry into and out of the gastight area will be noted.

The overall building leak rate will be determined over a 40-hour test period. The results will be corrected for wind velocity and direction, barometric pressure, and temperature conditions used in the calculations of building leak rate and compared to values used in the hazards analysis. Sample calculations are shown in Appendix D.

The building leak rate test will verify the design and construction of the ATR building. The continued integrity of the gastight area is based on administrative control and practices developed in operating test reactors. New penetrations will be sealed and individually leak tested. Inspection of seals for doors and other entry ports will be made under preventive maintenance procedures.

6. CONCLUSIONS

The ATR reactor building has been designed with particular emphasis on the gastight capability of the confined portions of the building. The leak rate determination test will confirm that the building is capable of containing the maximum release under the most adverse weather conditions without a dangerous radioactive release to on-site or off-site facilities. The continued integrity of the gastight area is under administrative control and based on proven procedures. A water spray is also provided to wash down the air and walls of the reactor room in the event of a dangerous release of fission products, especially iodine, in the reactor area. This fine mist spray system, while not automatic, may be turned on in the reactor control room or at the crash gate in the south boundary fence. This system will drain the fission products into the liquid waste tank for disposal.

III-3. REFERENCES

1. Safety and Fire Protection Design Criteria Manual, United States Atomic Energy Commission, Idaho Operations Office, Idaho, IDO-12008 (August 1959).
2. L. Baurmash et al, Leakage Characteristics of Openings for Reactor Housing Components, Atomics International, NAA-SR-Memo 5137 (June 20, 1960).
3. G. A. DeMarrais, The Engineering Climatology of the National Reactor Testing Station, United States Weather Bureau, Idaho Falls, Idaho, IDO-12004 (November 1958).

IV. REACTOR STRUCTURE, SHIELDING, AND EQUIPMENT

IV-1 Reactor Vessel

IV-2 Shielding

IV-3 Experimental Facilities

IV-4 References

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IV-1. REACTOR VESSEL

1. REACTOR VESSEL DESCRIPTION

1.1 General

The reactor vessel contains the core and reflector assembly and provides space for in-pile experimental facilities. Access ports are provided to facilitate core refueling, insertion, and removal of experiments and associated facilities. Sufficient vessel volume above the core is provided to permit the light water moderator-coolant to serve as an upper core shield during reactor operation and shutdown handling work. Additional design features include a manual refueling port, experiment and reactor instrument nozzles, and bottom- and top-head penetrations for various functions.

The basic vessel shell configuration is a 12-foot OD right circular cylinder, approximately 26.5 feet long. An elevation of the vessel is shown in Figure IV-1-1. The shell is closed at the bottom by a welded hemispherical head, and at the top by a flanged and bolted full-opening head of standard elliptical cross section. Both top and bottom heads have central openings closed by flat circular cover plates seated inside the vessel.

The inside height of the vessel between cover plates is about 32 feet. The total volume is 3250 cubic feet of which about 3000 cubic feet is water. The nuclear core, concentrically located within the vessel, is about 18 feet below the top head. Figure IV-1-2 shows a cutaway of the vessel and internals.

The basic coolant flow path, shown in Figure IV-1-1, is through two inlet nozzles located in the bottom head and upward in two annuli defined by the vessel wall, thermal shield, and the cylindrical inlet flow baffle. The two annuli end approximately three feet above the core where the inlet water becomes part of the water reservoir (1850 cubic feet) above the core. This reservoir feeds the coolant flow down through the core and reflector assembly into a lower plenum which is divided in quadrants by baffles. Four internal piping assemblies transport the coolant from the lower plenum four quadrants to four outlet nozzles located on the vessel approximately 3.5 feet above the top of the core.

1.2 Vessel Rating

The vessel, heads, nozzles, and attachments are designed for 390 psig pressure at a wall temperature of 240°F. Maximum operating coolant conditions are 370 psig at a fluid temperature contacting vessel walls of 130°F. Normal operating conditions are 355 psig and 125°F.

1.3 Materials of Construction

The reactor pressure vessel is constructed of Type 304 stainless steel using applicable ASME materials specifications. Table IV-1-1 shows the applicable specifications and associated physical properties.

To minimize the possibility of intergranular corrosion, the carbon content of the material contacting the primary coolant is restricted to 0.060 percent by weight maximum. This precaution was taken because the nitric acid addition to the primary coolant reduces the pH to a value of 5 [1].

① TOP HEAD CLOSURE PLATE	⑪ THERMAL SHIELD SUPPORT BRACKETS	⑳ INSTRUMENTED CAPEULE PENETRATION
② UPPER OVERFLOW NOZZLE	⑫ BOTTOM HEAD	㉑ SUPPORT SKIRT
③ AIR PURGE PORT	⑬ BOTTOM HEAD CLOSURE PLATE	㉒ COOLANT OUTLET NOZZLE
④ LOWER OVERFLOW NOZZLE	⑭ WELDED SEAL MEMBRANE	㉓ OUTER SHIM DRIVE NOZZLE
⑤ REFUELING PORT	⑮ NECK SHIM CONTROL DRIVE NOZZLES	㉔ EXPERIMENT BRACING "T" BARS
⑥ DROP TUBE COVER STORAGE BRACKET	⑯ TOP HEAD	㉕ VESSEL SHELL
⑦ NUCLEAR INSTRUMENT THIMBLE	⑰ SIPHON BREAK PIPE	㉖ SAFETY ROD DRIVE NOZZLE
⑧ DROP TUBE COVER	⑱ TOP HEAD CLOSURE PLATE SUPPORT	㉗ COOLANT INLET NOZZLE
⑨ DROP TUBE	㉙ PRESSURE TUBE PENETRATION	㉘ BOTTOM FLUSHING CONNECTION NOZZLE
⑩ DROP TUBE BLISTER	㉚ LOOP PIPING PENETRATION	㉛ PRESSURE TUBE PENETRATIONS

IV-1-1 ATR VESSEL ELEVATION.

FIG. IV-1-2 ATR VESSEL CUTAWAY.

1.4 Top Head, Cover Plate, and Seal Arrangement

The elliptical top head is bolted to the reactor vessel through flat faced flanges double grooved to contain two natural rubber O-rings. The center circular cover plate, recessed and bolted to the top head, mates with the top head by flat face flanges double grooved to contain two natural rubber O-rings. Internal vessel brackets hold the cover plate in place when the top head is removed. The five elliptical refueling ports and covers, equally spaced around the center cover plate, are made with flat face flanges containing a single groove to hold a natural rubber O-ring. The grooves in the five refueling port cover flanges are specially shaped to capture the O-ring during handling.

1.5 Bottom Head, Cover Plate, and Seal Arrangement

The hemispherical bottom head is welded to the reactor vessel. The

TABLE IV-1-1

MINIMUM TENSILE PROPERTIES -- ROOM TEMPERATURE

ASME Specification	Form	Tensile Strength (psi)	Yield Strength (psi)	Elongation in 2 Inches (%)
SA-240	Plate	75,000	30,000	40
SA-182	Forgings	75,000	30,000	45
SA-336	Forgings	70,000	30,000	30
SA-376	Pipe	70,000	30,000	35
SA-213	Tubing	75,000	30,000	35

bottom-head closure plate is seated on an offset of the bottom-head flange. The closure plate is grooved to contain a metal O-ring on the flat seating surface. In addition to the O-ring seal, a removable welded membrane provides zero leakage from the bottom head and closure plate to the subpile room.

1.6 Penetrations

In addition to the top- and bottom-head closure plates, the ATR vessel has 58 major penetrations and 44 major nozzles. All openings in the basic shell for these penetrations and nozzles are reinforced in accordance with the ASME Code provisions. Some of these openings have received individual reinforcement. Thicker vessel sections, based on ligament efficiencies, are provided in areas containing groups of penetrations such as loop piping penetration and the drop tube blister. Table IV-1-2 presents a list, size, purpose, and location of major penetrations and nozzles. Other small nozzles and penetrations, above and below the core, are required for venting, seal water entry, and N-16 monitoring lines. These openings (1/8 to 1-1/2 inch ID) are in accordance with the ASME Code provisions.

TABLE IV-1-2
REACTOR VESSEL NOZZLES AND PENETRATIONS

<u>NOZZLES</u>				
<u>Quantity</u>	<u>Nominal ID (inches)</u>	<u>Purpose</u>	<u>Location in Vessel</u>	
2	23	Coolant inlet	Bottom head, 12 ft below the core	
4	15	Coolant outlet	Shell course, 3.5 ft above the core	
8	6	Safety drive line	Shell course, 9.5 ft below the core	
8	3	Outer shim drive line	Shell course, 1 ft above the core	
4	2	Neck shim drive line	Bottom closure plate	
4	2	Fire-water inlet	Bottom closure plate	
1	8	Upper overflow	Shell course, 14.5 ft above the core	
1	8	Lower overflow	Shell course, 10 ft above the core	
12	5	Instrument thimbles	Shell course, 13 ft above the core	

<u>PENETRATIONS</u>				
<u>Quantity</u>	<u>Nominal ID (inches)</u>	<u>Purpose</u>	<u>Method of Closure</u>	<u>Location in Vessel</u>
5	20 x 40 elliptical	Refueling port	Bolted cover with O-ring	Top head
5	9-3/4	Air purge port	Bolted cover with gasket	In-refueling port cover
9	6	Pressure tube penetration	Packing gland	Top closure plate
9	5-1/2	Pressure tube penetration	Packing gland	Bottom closure plate
4	12	Loop piping	Bolted cover with gasket	Shell course, 14 ft above the core
9	10	Loop piping	Bolted cover with gasket	Shell course, 13 ft above the core
16	3	Capsule instrument leads	Bolted cover with gasket	Shell course, 12 ft above the core
1	16	Drop tube port	Internal cap with O-ring	Blister, 1/2 ft above the core

1.7 Attachments

The only direct external vessel wall attachments are the welded clamps for the support skirt.

The internal vessel attachments consist of 12 thermal shield support brackets welded to the shell course approximately two feet above the bottom-head attachment weld, vertically mounted T-bars welded on the shell course above the core to provide support brackets for experiments, and three extension arms to support the top closure plate when the full-opening head is removed. A two-

inch-thick internal ring for experiment supports is also provided approximately 3-1/2 feet below the top of the vessel shell.

1.8 Vessel Support

The ATR vessel is supported by a conical skirt mounted to the exterior wall of the vessel approximately 10-1/2 feet above the top of the core. This skirt extends outward to a base ring bolted to the structural concrete foundation.

1.9 Flushing, Venting, and Overflow

Four flushing nozzles are provided on the bottom-head closure plate to provide a method to reduce radioactive crud accumulation on the top of this plate. The flush lines also provide a positive pressure differential between the flush water and reactor coolant allowing clean water to leak past the metal O-ring seal into the reactor, preventing crud deposit accumulation in the area between the seal weld membrane and the O-ring seal.

The five refueling port covers are each equipped with smaller ports containing bolted covers with gaskets. These ports serve to vent the reactor vessel as required.

Two overflow nozzles are provided to reduce and maintain the reactor water level during shutdown to simplify maintenance and experiment handling.

Four one inch siphon break pipes extend from each of the four outlet pipes to maintain water level above the fuel in the event of certain ruptures in the primary system.

2. DESIGN CRITERIA

The vessel, its attachments, and penetrations have been designed and constructed to meet all requirements of the ASME Unfired Pressure Vessel Code and the requirements of all code cases and interpretations applicable to pressure vessels for nuclear service.

2.1 Applicable Codes

The completed vessel is approved under the provisions for nuclear vessels as stated in ASME Code Case 1270N-3 and has an ASME Code stamp. The reactor vessel was inspected in accordance with the ASME Boiler and Pressure Vessel Codes. The ASME reference documents are as follows:

Section II - Materials Specifications

Section VIII - Unfired Pressure Vessels

Section IX - Welding Qualifications

1962 Edition and interpretations of ASME Code including Cases 1270N-5, 1271N, 1272N-5, 1273N-7, 1274N-6, and 1275N.

Materials used in the reactor vessel and the tests performed were as follows:

- (A) Stainless Steel Plate and Sheet ASME SA-240, Type 304 0.06 Max Carbon
 - (1) Ultrasonic inspection for all materials used for pressure containment.
- (B) Stainless Steel Forgings ASME SA-182, Grade F, Type 304 or ASME SA-336, Class F-8.
 - (1) Macro-etch test.
 - (2) Ultrasonic inspection for all materials used for pressure containment.
 - (3) Carbon content procedure in accordance with ASTM E-30.
- (C) Stainless Steel Seamless Pipe ASME SA-376, Type 304.
 - (1) Carbon content procedure in accordance with ASTM E-30.
 - (2) Supplementary Paragraphs S-2 through S-5.
 - (3) Ultrasonic and liquid penetrant inspection for all materials used for pressure containment.
- (D) Stainless Steel Tubing ASME SA-213, Type 304
 - (1) Carbon content procedure in accordance with ASTM E-30.
 - (2) Ultrasonic and liquid penetrant inspection for all materials used for pressure containment.
- (E) Stainless Steel Welding Electrode ASME SA-298 or ASME SA-371
 - (1) Type E-308 ELC except carbon limited to 0.060 weight percent maximum.
 - (2) Met requirements of SA-298, Section 7.
- (F) Principal Bolting Materials
 - (1) Met ASME Specification for pressure, temperature conditions, and locations.
 - (2) Ultrasonic inspected.
 - (3) 100 percent liquid penetrant inspected.

The following inspection requirements over and above the ASME Code requirements were imposed on all vessel welds.

Completed Weld

Liquid penetrant inspection
on both sides of weld

Adjoining surfaces of the
base metal prior to welding

Liquid penetrant inspection

After first weld pass

Liquid penetrant inspection
on both sides of weld

After one-half weld thick-
ness

Liquid penetrant inspection

Procedures for making all repairs to the vessel during fabrication were submitted to the A. E. for specific approval.

2.2 Stress Analysis

An ASME Code Analysis [2] was made for the ATR vessel including top and bottom head closure plates, nozzles, and penetrations. Figure IV-1-3 shows the location of the regions analyzed. A summary of the maximum stresses (combined) for principle locations together with allowable values are listed in Table IV-1-3.

The thickness and areas of reinforcement required for all vessel nozzles and openings have been calculated. Several major nozzle requirements together with vessel design values are shown in Table IV-1-4.

Calculations of all bolt requirements have been made.

The loads imposed on the reactor vessel and nozzles were included in calculations determining nozzle reinforcement. The bending moments were computed for the rod drive nozzles. Piping reactions for all nozzles were computed and resulting loads used in calculating required nozzle reinforcement.

The results of the stress calculations including nozzle reinforcement, weld areas, bolting pipe, and other external reactions show that the ATR vessel design meets all code requirements.

2.3 Cyclic Behavior

The ATR vessel has been designed to accomodate rapid thermal cycling

FIG. IV-1-3 STRESS ANALYSIS LOCATIONS.

TABLE IV-1-3

ASME CODE STRESS ANALYSIS SUMMARY

Location	Material	Design Temperature (°F)	Design Pressure (psi)	Location Number (See Fig. IV-1-3)	Thickness (inches)	Allowable Stress (psi)	Combined Stress (psi)
Cylindrical Shell	SA 240 Type 304 Stainless Steel	240	390	2.1	2	15990	14744
Bottom Hemispherical Head	SA 240 Type 304 Stainless Steel	240	390	2.3	2	15990	7724
Top-Head Flange	SA 182 Type 304 Stainless Steel	240	390	2.4	16	23985 15990 15990	10026 Hub 6329 Tan 5807 Rad
Vessel Flange	SA 182 Type 304 Stainless Steel	240	390	2.5	15.75	23985 15990 15990	8610 Hub 5276 Rad 5638 Tan
Cylindrical Shell	SA 240 Type 304 Stainless Steel	240	390	2.7	5	15990	6022

TABLE IV-1-4

ATR VESSEL NOZZLE DESIGN

Nozzle Description and Location Number (See Fig. IV-1-3)	Required Thickness (inches)	Design Thickness (inches)	Required Area of Reinforcement (in ²)	Design Area of Reinforcement (in ²)
Top Head Central & Refueling 2.8; 2.9	Shell 1.505, Nozzle 1.5	Shell 4.875, Nozzle 5.0	87.29	218.64
Bottom-Head Flange -- 2.12	Shell 0.856, Nozzle 0.749	Shell 2.0, Nozzle 6.23	51.79	124.22
Inlet Nozzle -- 2.13	Shell 0.856, Nozzle 0.288	Shell 2.0, Nozzle 1.25	19.9	32.61
Safety-Rod Drive Nozzle -- 2.14	Shell 1.733, Nozzle 0.084	Shell 2.0, Nozzle 0.5	11.638	12.79
Drop Tube -- 2.15	Shell 0.391, Nozzle 0.198	Shell 2.0, Nozzle 1.25	6.256	32.32

caused by fast start-up and shutdown emergencies. Analyses based on stress calculations are shown below.

Allowable Cycles (No.)	Temperature range of Transient (°F)	Time of Transient (sec)
2000	120 to 170	45
2000	170 to 120	3

2.4 Radiation Damage Analysis

The maximum fast neutron flux (> 1 MeV) at the reactor vessel wall will be 1×10^7 n/cm²-sec yielding a total dosage for the 20 year vessel design life of 1×10^{16} n/cm². Irradiation damage data for austenitic stainless steels with total fast flux dosages above 5×10^{21} n/cm² show Charpy impact values of about 65 foot-pounds at 80°F and 20 foot-pounds at -320°F. These values show considerable ductility at doses $> 10^5$ times the value expected for the ATR Vessel.

2.5 Fabrication

Fabrication and assembly procedures including welding were reviewed and approved by the ATR design contractors prior to commencing vessel manufacture. Also included were major tests, inspections, and weld examinations standards. Repair work to vessel required specific approval. Certified records for all inspections were provided. All mill test reports, chemical analyses, and thermal treatment histories of vessel components were also provided.

ASME Code qualified welders performed all vessel welding. Electrode material provided a deposit containing a maximum carbon content of 0.060 weight percent in accordance with SA-298 Type E 308 ELC-15. All pressure containment welds were liquid penetrant inspected on both sides after the first pass, after completing 1/2 the weld thickness, and the completed weld. The completed welds were radiographed in accordance with paragraph UW-51, Section VIII of the ASME Unfired Pressure Vessel Code and ASTM designation E-94-52T.

2.6 Corrosion

Corrosion behavior of Type 304 stainless steel was examined under coolant conditions expected during ATR operation. The tests were run for a maximum of 4000 hours. Corrosion rates were negligible (<1 mil/year). Tests included welded specimens, specially prepared strips with prior intergranular attack, and stressed specimens. Metallographic examination of control and test specimens did not reveal any propagation of the intergranular attack [10].

3. VESSEL INTERNALS

3.1 Description of Vessel Internals

The centrally located core support assembly is a stacked group of short cylindrical tank sections. This assembly includes a core support tank, flow distribution tank, reflector support tank, and core-reflector tank as shown in Figures IV-1-4, -5, -6, and -7. The 14.5-foot-high assembly has a 51-inch basic ID and is flanged and bolted to the reactor vessel bottom closure plate.

The core support tank is a vertical cylindrical section five feet high equipped with top and bottom flanges and end plates. The bottom flange is bolted to the vessel bottom closure plate. Tubular columns extending up and outward from the tank wall and bottom flange help support the larger diameter top flange. The top flange supports the conical transition section of the inlet flow baffle and the flow distribution tank. The core support tank contains eight vertical guide tubes for the safety rod drives and experiment tubes located between the top and bottom end plates. It also contains the N-16 monitoring tube assemblies. Six inches below the top end plate, eight horizontal bearing housings penetrate the core support tank wall engaging the eight safety rod drive guide tubes. The guide tubes and bearing housings contain, align, and position the safety rod drive shafts. The core support tank, guide tubes, and bearing housings are fabricated of stainless steel.

The flow distribution tank is a vertical, cylindrical section 3-1/2 feet high. The tank is closed on the bottom with a flat end plate forming a mating flange

① ATTACHMENT TO NECK SHIM HOUSING	⑭ INSTRUMENT LEADS
② FLUX TRAP BAFFLE	⑮ SAFETY ROD ASSEMBLY SUPPORT POINT
③ OUTER CAPSULE IRRADIATION FACILITY (ALUMINUM)	⑯ PRESSURE TUBE ASSEMBLY
④ CANAL LINER	⑰ THERMAL EXPANSION JOINT
⑤ DROP TUBE	⑱ CORE-REFLECTOR TANK (ALUMINUM)
⑥ OUTER REFLECTOR BLOCKS (ALUMINUM)	⑲ PROTECTIVE COVERS
⑦ THERMAL SHIELD	⑳ OUTER SHIM DRIVE NOZZLE
⑧ INLET FLOW BAFFLE	㉑ OUTLET FLOW PIPE
⑨ SAFETY ROD CAVITY	㉒ REFLECTOR SUPPORT TANK (ALUMINUM)
⑩ N-16 SAMPLE TUBE PENETRATION	㉓ FLOW DISTRIBUTION TANK (STAINLESS STEEL)
⑪ BERYLLIUM REFLECTOR SUPPORT POINT	㉔ FLUX TRAP COOLANT PLENUM
⑫ FUEL ELEMENT SUPPORT POINT	㉕ SAFETY ROD DRIVE NOZZLE
⑬ OUTER SHIM CONTROL CYLINDER BEARING SUPPORT	㉖ CORE SUPPORT TANK (STAINLESS STEEL)

FIG. IV-1-4 VESSEL INTERNALS ELEVATION.

FIG. IV-1-5 VESSEL INTERNAL CUTAWAY.

with the core support tank top flange. The tank is open at the top with an external flange attached to the shell. The eight safety rod drive tubes in the core support tank have three-foot joining tubular extensions welded rigidly to the bottom end plate of the flow distribution tank. The safety rod drive housing top guides rest on the top of these tube extensions. These tubes also support and locate the flux trap baffles. The removable neck shim rod housing is supported by a stub tube located around the center penetration of the end plate. A special center flux trap baffle assembly extends down through the flow distribution tank to the bottom-head closure plate. A plenum extends one foot above the bottom end plate of the flow distribution tank. This plenum receives the coolant flow from the eight flux trap areas and divides it equally to the four quadrants of the flow distribution tank.

The flow from the center flux trap is discharged directly to the flow distribution tank. Four radial baffles, extending from the top of the plenum to the bottom of the reflector support tank, divide the flow distribution tank into quadrants. Each quadrant has a flanged opening in the tank wall mating with a flange from one of the four outlet coolant flow pipes. Additional coolant lines from the outer aluminum reflector capsule facilities enter the flow distribution tank lower plenum area. The flow distribution tank, its tubular extensions, and bottom end plate are constructed of stainless steel.

The reflector support tank is a cylindrical casting one foot high, flanged on top and bottom. The bottom of the beryllium blocks and bottom ends of inner capsule irradiation cans are located by metal cups in the reflector support tank. The tank is constructed of aluminum and bolted to the top flange of the flow distribution tank.

The core reflector tank, a vertical, cylindrical section approximately five feet high, is open at the top and bottom. This tank, constructed of aluminum, is attached to the top flange of the reflector support tank to serve primarily as a coolant container for the coolant flowing down through the core-reflector assembly and as the support member for the outer shim control drive gear box beams which support the beryllium cylinders and locate the top ends of the beryllium reflector blocks. It also serves as an outer physical boundary to maintain the core reflector components in their proper radial orientation. An internal flange at the bottom supports the beryllium reflector blocks. The top rim of the tank contains bolt holes for attaching instrumented capsule supports. The core reflector tank extends approximately seven inches above the top of the active

① BERYLLIUM REFLECTOR BLOCK	⑭ FUEL STORAGE RACK
② PRESSURE TUBE ASSEMBLY	⑮ CORE REFLECTOR TANK
③ FUEL ELEMENT	⑯ OUTER SHIM CONTROL CYLINDER
④ OUTER SHIM DRIVE SHAFT	⑰ INLET FLOW BAFFLE ASSEMBLY
⑤ REACTOR VESSEL	⑱ N-16 SAMPLE TUBE
⑥ THERMAL SHIELD	⑲ GEAR BOX SUPPORT BEAM
⑦ INSTRUMENT THIMBLE	⑳ OUTER CAPSULE IRRADIATION FACILITY
⑧ NECK SHIM ROD HOUSING	㉑ SAFETY ROD
⑨ INNER CAPSULE IRRADIATION FACILITY	㉒ DROP TUBE
⑩ PRESSURE TUBE LOCATOR TIES	㉓ REFUELING PORT RP-1
⑪ OUTER SHIM GEAR BOX	㉔ REFUELING PORT RP-2
⑫ OUTLET FLOW PIPE	㉕ REFUELING PORT RP-3
⑬ OUTER FLUX TRAP BAFFLE
HOLD DOWN LATCH	㉖ REFUELING PORT RP-4
	㉗ REFUELING PORT RP-5

FIG. IV-1-6 VESSEL INTERNALS PLAN.

① PRESSURE TUBE LOCATOR TIES	⑫ CENTER FLUX TRAP BAFFLE
② NECK SHIM ROD HOUSING	⑬ PRESSURE TUBE ASSEMBLY
③ OUTER SHIM GEAR BOX	⑭ SAFETY ROD ASSEMBLY (UP POSITION)
④ UPPER REFLECTOR ALIGNMENT CUPS	⑮ OUTER SHIM GEAR BOX SUPPORT BEAM
⑤ SAFETY ROD ASSEMBLY (DOWN POSITION)	⑯ BERYLLIUM REFLECTOR BLOCKS
⑥ INNER CAPSULE IRRADIATION FACILITY	⑰ CORE-REFLECTOR TANK
⑦ OUTER SHIM CONTROL CYLINDER	⑱ N-16 SAMPLE TUBE CAVITY
⑧ LOWER REFLECTOR ALIGNMENT CUPS	⑲ FUEL ELEMENT
⑨ POSITION INDICATOR LINE	⑳ FLUX TRAP BAFFLE
⑩ FLOW DISTRIBUTION TANK	㉑ REFLECTOR SUPPORT TANK
⑪ CAPSULE FACILITIES ASSEMBLY	㉒ FUEL ELEMENT SIDE PLATE SUPPORT

FIG. IV-1-7 CORE ELEVATION.

fuel in the core and keeps the fuel covered with water in the event the remainder of the vessel is drained.

The inlet flow baffle, a vertical, right cylindrical section approximately 10 feet in diameter, has a flanged bottom, conical transition section attached to a support flange on the core support tank (Figure IV-1-4). This baffle assembly directs most of the coolant flow entering the bottom of the vessel up through the annuli formed by the baffle, thermal shield, and the vessel wall to the upper portion of the vessel. The upper cylindrical portion of the baffle serves as the inner thermal shield. The baffle is slotted in the vicinity of the drop tube blister. The inlet flow baffle assembly is constructed of stainless steel. A special thermal shield section is constructed of stainless steel and is attached to the inlet baffle cylinder through a flanged and bolted section at their intersection.

The ATR thermal shield assembly reduces the gamma heating in the concrete walls surrounding the vessel shell, but it is not specifically required to protect the vessel walls. The thermal shield is a vertical, right circular cylinder approximately 11 feet in diameter, 10 feet high, and 2.5 inches thick. Located in the annulus between the inlet flow baffle cylinder and the vessel shell, it is supported by 12 gussets on the vessel wall and is positioned vertically with its midpoint even with the core midplane. It is cut out in the vicinity of the drop tube and is cooled on both sides by the main coolant inlet flow entering the reactor. The thermal shield cylinder, constructed of stainless steel, was fabricated in four sections and assembled and bolted together in the vessel. Mounting brackets used to attach the thermal shield cylinder to its support gussets maintain the cylinder concentric to the vessel shell.

The primary coolant leaving the core through the flow distribution tank is carried to the vessel outlet nozzles through four outlet flow piping assemblies, one for each core quadrant (Figures IV-1-4 and -5). These outlet flow piping assemblies consist of 16-inch OD pipes flanged at both ends so they may be attached to the tank and the outlet nozzles. The flange attachment on the vessel outlet nozzles facilitates the disassembly from above. A piston-ring sealed, sliding expansion joint is installed below the upper elbow of these assemblies to provide piping assembly flexibility and to accommodate vertical temperature and pressure motion between elbows. All piping assemblies are constructed of stainless steel except for the 17-4 PH piston rings. Each of the nozzles for these piping assemblies has a one-inch siphon breaker pipe terminating just below the reactor vessel top head. These pipes bypass a small amount of primary coolant directly from the vessel top to the outlet nozzles and keep the core covered with water should the water level in the vessel drop.

Besides the experimental loop facilities and the capsule test facilities in the beryllium reflector area, the ATR provides two outer capsule irradiation facilities (Figure IV-1-4). These capsule facilities consist of aluminum blocks, approximately four inches by four inches by five feet long, located and supported in rectangular water tanks attached to the outer wall of the core reflector tank at two diametrically opposed positions. The top of these tanks is open to facilitate the handling of experiments. The bottom of the tank furnishes an alignment and support grid for the blocks and a plenum for coolant flow down through the blocks. The coolant is taken from this plenum via a four-inch pipe to the plenum at the bottom of the flow distribution tank. Ten capsule positions are provided for the side of the core reflector tank facing the drop tube, and 20 positions are on the opposite side. The flow containment tanks and the support grids are constructed

of stainless steel. Plugs are provided for flow restriction in unoccupied capsule and specimen positions.

An internal fuel element storage facility in the ATR vessel is provided for the convenience of the operator to store irradiated fuel elements reducing down time under specific operational requirements. Two storage racks, each accommodating 12 fuel elements, are located in the annular water space between the core reflector tank and the inlet flow baffle cylinder in diametrically opposed positions (Figure IV-1-6). Each rack consists of three individual box-type structures located and supported within a coolant tank. An additional box-type structure accommodating one element is provided in the coolant tank but is used in handling operations only. This tank is supported by gussets from the inlet flow baffle cylinder. The "egg-crate" box containers are for the elements and have nuclear poison material surrounding each individual element cell. The k_{∞} of this arrangement is approximately 0.7. The coolant tank allows the coolant to flow down the outside of the fuel element storage rack and then up through the stored elements by natural circulation. Since the tank has no bottom coolant connections, it maintains water over the stored elements even if the vessel is drained. A weir arrangement from the core reflector tank allows water to overflow to each of the fuel element storage tanks when the core reflector tank level is maintained by emergency and/or flush water flow. The coolant tank is constructed of stainless steel; the box-type structures (baskets) are of cadmium encased in aluminum.

The additional gamma heating in the inlet flow baffle adjacent to these storage facilities is approximately 50 percent above the basic heating rate for the balance of the baffle cylinder. The resulting thermal stresses (< 2000 psi) are well below the allowable limits for the structure.

Nine flux traps for experimental loop positions are provided by the ATR core configuration (Figure IV-1-6). The outer boundary of these spaces, except the center one, is defined by vertical tubes (flux trap baffles) extending through the core area. These baffles, 5-1/4-inch ID and 5-7/8-inch OD, separate the coolant flow through the fuel element from that flowing through the test areas. They also support and index the fuel elements and physically define the outer limit of the test spaces. The flux trap baffles extend from inside the support tubes on the core support plate to approximately seven inches above the core fuel region. An integral flange near the bottom rests and indexes on the top of the core support tubes. This slotted-top flange orients and supports the fuel elements. The top edge of the flux trap baffles is slotted to orient the top of the fuel elements. The four inner flux trap baffles in the corner positions are fitted with a vertical aluminum strip approximately 4.5 feet long on the portion of their periphery not bounded by fuel elements. This strip supports a tongue and groove alignment joint for the neck shim housing. The neck shim rod housing provides hold-down for the four inner flux trap baffles. The four outer flux trap baffles are held down and aligned by a connection with the gear box support plate for the outer shim control cylinder drives. The center flux trap is bounded by a special center-post flux trap baffle mounted on the core support plate. The center flux baffle is inserted through a central opening in the neck shim rod housing. All flux trap baffles are constructed of aluminum in the fueled area. The upper fuel support area of the baffle is Inconel X and the lower area, stainless steel.

The neck shim rod housing is approximately 13 feet long, extending from the bottom end plate of the flow distribution tank up through the core area and terminates approximately four feet above the active fuel region. The housing is cruciform in shape with four radial extension arms joining the four corner flux

trap baffles. It is constructed of two vertical layers to facilitate installation and removal from the reactor. Each radial extension has six in-line circular channels through the neck region to guide the neck shim rods. A bottom eight-inch-high transition section converts the channels of each arm from the in-line pattern to a circular pattern to form a connection with the neck shim rod drive housing. Except for top cover plates, the housing is assembled completely before it is placed in the reactor. The housing is constructed of aluminum except for stainless steel lower transition sections.

The nine experimental pressure tube assemblies are located and guided in place by a tie system which is an integral part of the top cover plates on the neck shim rod housing. Dummy loops are provided for positions not occupied with actual experiment loops.

3.2 Assembly Mock-Up

The design of the vessel internals and the associated handling procedures were developed in conjunction with the use of a full scale reactor mock-up. This mock-up also included the subpile room and sufficient headroom above the vessel top head to demonstrate use of the handling tools.

IV-2. SHIELDING

1. GENERAL DESCRIPTION

Shielding calculations for the ATR are based on techniques and procedures in general use in the industry. The equations used for flux calculations from regular geometric sources were based on work by Sievert [3] and extended by others [4, 5, 6], and were programmed for machine computations. A program was developed for computing the secondary gamma-ray flux through a laminated semi-infinite shield region. Heat generation rates were determined by computing the primary and secondary gamma fluxes using a point-source summation program. Source activities during operation include fast and thermal neutron flux inside the reactor computed by 20 group diffusion equations. The fluxes are developed as a function of space and lethargy. The thermal flux values when used for secondary gamma calculations were checked with good agreement with Lid Tank Shield Facility data. All gamma-ray source strength and attenuation calculations had unit energy intervals and the appropriate form of buildup for dose rate or heat generation rate. Details of the calculations have been reported [7].

2. REACTOR VESSEL

The ATR vessel shielding provides biological protection and prevents excessive structural heating caused by radiative absorption.

2.1 Internal Shielding

Internal shielding in the ATR vessel to reduce structural heating consists of water and thermal shields. Attenuation of fast neutrons is rapid outside the core region, and the contribution to structural heating in the vessel walls and biological shields is therefore negligible. Gamma rays provide the primary heat source outside the core. Two stainless steel thermal shields surround the core and reduce the gamma heating rate in the pressure vessel and concrete shield surrounding the pressure vessel. The inner thermal shield, ID of 118 inches, is 1.5 inches thick and also serves as the inlet coolant flow baffle. The outer thermal shield, ID of 128 inches, is 2.5 inches thick. The arrangement of the thermal shields is shown in Figures IV-1-2 and -5.

2.2 Nozzle Trench

The shielding around the vessel in the nozzle trench area is designed to restrict the general radiation level in the trench (neglecting streaming through the nozzle penetrations) from elements in the core and temporary storage racks to 12.5 mrem/hr or less, three hours after shutdown. No access is allowed in the nozzle trench during operation. The shielding consists of the two-inch-thick vessel support skirt which surrounds the pressure vessel and extends from the floor of the capsule trench to the four-inch vessel wall section. Although shield plugs will be provided in the instrument wells, some streaming of radiation may be expected during removal or insertion of instruments. Access to the nozzle trench is through a single door containing a lock.

2.3 Top-Head Shield

Figure IV-3-5 shows the arrangement of the vessel top-head shield. The working platform is 12 inches thick and is designed to restrict the dose rate at its upper surface to 1.0 mrem/hr or less during normal full power operation. Its thickness is governed by the dose rates from the N-16 activity in the vessel upper plenum. The shield cylinder is sized to restrict the transient dose rate during specimen transfer to 1 rem/hr or less at the shield surface. The cylinder shielding is composed of three inches of steel and nine inches of lead. The design is based on the removal of a test specimen after one hour shutdown following infinite operation at 800 kW. During operation, the dose rate at the shield cylinder surface is computed to be less than 1.0 mrem/hr. The design of the loop transfer shield plate is based on the same criteria as those for the shield cylinder, and results in a shielding thickness of 15 inches of steel. The purpose of the top-head ring shield is to prevent radiation streaming through the top head and scattering from the working platform and concrete floor. The shield is composed of four inches of lead, two inches of steel, and is designed to reduce the dose rate around the top-head area to less than 1.0 mrem/hr during normal full power operation.

2.4 Primary Biological Shield

The reactor vessel is surrounded by a composite shield which includes three feet of high-density (about 220 lb/ft³) magnetite concrete, two feet six inches of structural concrete and five feet six inches of ordinary concrete (Figures III-1-4 and -8). The concrete completely surrounds the vessel except in the area shielded by the working canal water. The magnetite concrete extends 12.5 feet below, and 8.5 feet above the core midplane. This shield serves a dual purpose. During operation, this concrete acts as a thermal shield to protect the structural concrete from gamma heating. Heating rates in the magnetite concrete at core midplane elevation range from 4.8×10^{-4} watts/cc at the inner surface to 4×10^{-8} watts/cc at the outer surface. After shutdown it aids in shielding maintenance personnel in the piping corridor from fission products in the core and activated components within the vessel. One hour after shutdown, the dose rate in the piping corridor from sources within the reactor vessel is calculated to be less than 1 mrem/hr. The structural concrete extends from the floor of the subpile room to 10.5 feet above the core midplane, where it forms the floor of the capsule nozzle trench. During operation, no access is allowed in the capsule nozzle trench, and a three foot thickness of magnetite concrete shielding separates this area from the reactor floor area.

The pipe corridor, which contains the primary and test loop piping separates the 2.5-foot-thick and 5.5-foot-thick sections of the primary shield. The thickness of the 5-foot 6-inch ordinary concrete shield is governed by the N-16 activity in the primary piping. At full power, the dose rate from N-16 at the outer surface of this shield is calculated to be 6 mrem/hr. The dose rate from the operating core is negligible.

3. PRIMARY SYSTEM SHIELDING

3.1 Coolant Loop Shielding

The primary coolant piping is shielded by a minimum of five feet six inches of ordinary concrete. The shield thickness is based on N-16 activity in the

primary coolant. The maximum dose rate at the outer surface of this shield is calculated to be 6 mrem/hr. Similarly, the walls around the heat exchanger room are five feet six inches thick with a maximum dose rate of 6 mrem/hr at the outer shield surface. The floor above the heat exchanger room is six feet of ordinary concrete. The maximum dose rate at the floor surface is computed to be 1.5 mrem/hr.

The wall between the pump cubicles and heat exchanger area is five feet six inches of ordinary concrete. This reduces the radiation level from sources in the heat exchanger room to 2.5 mrem/hr in the pump cubicle. The walls between the pump cubicles are four feet thick and the calculated dose rate in any cubicle from an adjacent cubicle is 7 mrem/hr. The exterior walls of the pump cubicles are also four feet thick with a resulting dose rate of 7 mrem/hr in the motor area.

4. SUBPILE AND ROD ACCESS ROOMS

The subpile room is shielded from the core by 10 feet of water in the reactor vessel and by the one-foot-thick vessel bottom head. One hour after shutdown, the dose rate from the core through this shielding is computed to be less than 1 mrem/hr. Locally, higher radiation levels can be expected from streaming through loop penetrations in the bottom head. The wall between the working canal and the subpile room consists of four feet of ordinary concrete. This wall is designed to limit the dose rate from a spent fuel element stored in the deep end of the canal to 7.5 mrem/hr in the subpile room. The steel door in the subpile room is 10 inches thick and is sized for a dose rate outside the door of 2.5 mrem/hr from N-16 activity in the test loops. The average dose rate from test loop fission products is 3 mrem/hr during the first hour after a meltdown incident.

The floor between the subpile and rod access rooms is a combination of magnetite concrete, steel, lead, and polyethylene. Three feet of concrete, four inches of steel, and six inches of polyethylene shield the rod access room from core radiation and from N-16 in the test loops. Dose rates from these sources were computed to be less than 2 mrem/hr in the rod access room during full power operation. Where the test loop traps and exit piping are imbedded in the subpile room floor, the concrete is replaced by 6.5 inches of lead below the piping and 7.5 inches below the trap. The polyethylene is required below the lead and steel to ensure adequate attenuation of the neutrons from N-17 in the test loop coolant. The magnetite concrete wall between the piping corridor and the rod access room is three feet eight inches thick. The maximum dose rate in the rod access room from N-16 in the primary coolant is 8 mrem/hr. The total dose rate from all sources is less than 10 mrem/hr.

For the case of accidental release of fission products to a test loop, it was assumed that the fission products were dispersed throughout the test loop coolant volume. The resulting dose rates in the rod access room were negligible. It was also assumed that 10 percent of the activity released to the loop was immediately concentrated in the loop trap. With the 7-1/2-inch lead shield below the trap, the first-hour integrated dose in the rod access room was calculated to be 20 mrem.

5. CANAL SHIELDING

The ATR canal shielding is designed to protect personnel in areas adjacent to the various walls and canal bottom. Working areas above the canal are shielded by water in the canal. Fuel elements following discharge are normally placed in a storage rack located in the working canal for a period of about 15 days before transfer to the storage canal. The calculations for the walls and floor of the storage canal are based on a row of 40 irradiated (17 days) fuel elements, 4-inch center-to-center spacing, 15 days after shutdown, and 6 inches from both the canal wall and bottom. The canal bottom below the fuel element storage racks is seven feet of ordinary concrete. The dose rate below this floor is less than 1.0 mrem/hr. Where the canal floor passes over the experimental cubicles, 5.5 feet of ordinary concrete is used leading to a dose rate of 5.0 mrem/hr in the cubicles. Seven feet of ordinary concrete is provided for the lower portion of the canal walls, with resulting dose rates at the outer surface of the shields less than 1.0 mrem/hr.

Shielding for the ATR critical facility has been reported [8].

6. EXPERIMENTAL LOOP CUBICLES

The exterior walls around the test cubicles are constructed of two feet eight inches of magnetite concrete blocks and are designed to:

- (1) Attenuate radiation from N-16 in the test loops to 2.5 mrem/hr at the outer surface of the wall.
- (2) Restrict the integrated dose from fission products during the first hour following a test failure to 12.5 mrem at the outer surface of the wall.

The walls and floors between cubicles are sized for a dose rate of 2.5 mrem/hr inside a cubicle (with test loop shutdown) from an adjacent operating cubicle. The walls are constructed of magnetite concrete blocks and are two feet eight inches thick; the floors consist of four feet six inches of ordinary concrete. The cubicle ceilings in the first basement are made of ordinary concrete, five feet in thickness, in order to meet the 1.0 mrem/hr design level on the reactor floor.

The cubicle shielding was determined on the basis of N-16 activity in pressurized water test loops during normal full power operation. For the accident case, it was assumed that the fission products were distributed uniformly throughout the test loop coolant volume. The first-hour integrated dose outside the cubicle was computed to be less than the 12.5 mrem design level. A calculation was performed to determine the dose outside the shield wall from a concentration of fission products inside the cubicle. It was found that four percent of the activity released to the coolant in the postulated accident could be concentrated at a point in the cubicle without exceeding the 12.5 mrem limit.

7. DEMINERALIZER VAULTS

The bypass demineralizer vaults are located along the west wall on the north side of the reactor second basement. This area is outside the building boundaries defined by the grade level walls. The vaults are shielded by 3-1/2 feet of concrete block followed by a valve corridor of 3-1/2 feet and a final 8-inch concrete protective wall surrounding the valve corridor. The vaults are serviced through hatch covers in the first basement floor. The hatch covers in the floor above the vaults consist of 5-1/2 feet of concrete. The floor area above the vaults in the first basement is separated from the normal access areas by a concrete wall. The dose rates from the resin beds at the areas adjacent to the walls and ceiling of the vaults are variable and are, therefore, under administrative control.

8. SUMMARY OF DOSE RATES^[a]

8.1 Dose Rates from the Release of One ATR Fuel Element to the Primary System.

An estimate was made to determine the radiation levels outside the primary piping shielding from an accident involving release of fission products to the primary coolant. An accident was postulated whereby the gross fission product activity in one fuel element after 17 days operation was discharged to the coolant. It was assumed the fission products were dispersed uniformly throughout the coolant volume. The dose rate from a 36-inch pipe through a 5-foot 6-inch thick concrete shield, two minutes after the accident, was calculated to be 5 mrem/hr. Of all areas normally accessible during operation, the area outside the pump cubicles would have the highest radiation level due to the above 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 calculated to be 150 mrem/hr.

8.2 Radiation Control Areas

Shielding for certain areas has been provided for limited access during particular periods of the overall operation. It is uneconomical to provide shielding in these areas to cover all eventualities. These areas are placed under administrative control with respect to entry procedures and time of access. A list of these areas together with possible radiation sources are shown in Table IV-2-2.

[a] A summary of the calculated dose rates is given in Table IV-2-1.

TABLE IV-2-1

SUMMARY OF CALCULATED DOSE RATES

Location	Maximum Calculated Dose Rate During Full Power Operation (mrem/hr)	Maximum Calculated Dose Rate After Shutdown (mrem/hr)
Outside reactor concrete shield	6.0	< 1.0
Outside experiment H ₂ O loop cubicles (N-16 activity only)	2.5	< 1.0
Inside experiment H ₂ O loop	5.0 from adjacent operating cubicles (N-16 only); 6.0 from reactor and primary piping	< 1.0
Subpile room	No access	1 from core through vessel bottom head (does not include streaming through bottom-head penetrations)
Rod access room	10.0	< 2.5
Floor above heat exchangers	1.5	< 1.0
Reactor area, first floor	1.0	< 1.0 except during specimen and loop component transfer
Reactor vessel top head	1.0	< 1.0 except during specimen and loop component transfer < 25 during fuel transfer
Above storage canal and at floor level near parapet	1.0	1.0
Above storage canal during spent fuel transfer	50.0 transient	50.0 transient
Below storage canal	1.0	1.0
Outside walls of storage canal	1.0	1.0
Outside pipe tunnel	6.0	< 1.0
Motor area outside pump cubicles	7.0	< 1.0
Inside pump cubicles (Pump out of service)	4.0 from heat exchangers and piping, 7.0 from adjacent pump cubicle	
Nozzle access trench	No access	12.5 from core and stored fuel (does not include streaming through vessel penetrations)
Office areas	< 0.25	< 0.25 except during specimen and loop component transfer

TABLE IV-2-2

RADIATION CONTROL AREAS

Location	Operation	Radiation Source
Nozzle access trench	Removal of instruments and irradiated capsules	Streaming through instrument wells and capsule penetrations, activated capsules
Subpile room	Experiment loop removal, general maintenance	Contaminated experiment loops, streaming through in-pile loop penetrations, spent fuel elements passing through transfer chute
Pipe corridor	Maintenance on rod drive mechanisms	Contaminated experiment loops, spent fuel passing through transfer chute
Experiment loop test cubicles	Maintenance on test loop equipment	Contaminated test loops
Reactor area, first-floor	Removal of activated loop components	Streaming out of top and bottom of transfer cask
Reactor top-head area, top-head shield removal	Spent fuel removal general maintenance	Contaminated experiment loops, raised fuel elements
Top-head shield, and cask transfer station	Transfer of test specimen	Test specimen
Demineralizer Vault Areas	Change out of resin	Fission products and activated crud

9. HANDLING OPERATIONS

Handling operations in the ATR tank including refueling are performed manually. Specially designed tools of required length are manipulated through one of the elliptical ports in the top head. The water level in the vessel is maintained at one of two positions, determined by the particular operation and shielding requirements, by valves on the vessel overflow lines. The water provides the necessary shielding for personnel performing tank work. The various components are lifted from normal operating positions and transferred via the drop tube and canal transfer device to the working canal. Certain experiments, experiment facilities, and core equipment require special handling techniques.

The dose rates for the various handling operations were computed using source strengths based on material activation for the component being handled. The irradiation time used in the calculations was based on expected life of each component. All internals (except beryllium reflector blocks, neck shim housing, and fuel) can be handled 48 hours after shutdown with dose rates of less than 10 mrem/hr without additional shielding. The dose rate at the top of the reactor vessel during fuel transfer will be less than 25 mrem/hr when the element is at its maximum required elevation.

IV-3. EXPERIMENTAL FACILITIES

1. GENERAL DESCRIPTION

ATR experiment facilities consist of nine flux trap positions designed to provide high neutron flux over the entire core length to various types of dynamic loop experiments. In addition, inner capsule irradiation facilities, located within the core reflector tank, and two outer capsule irradiation facilities, located external to the core reflector tank, are provided. The experiment facilities arrangement is shown in Figure IV-1-6.

The nine dynamic loop positions are classified as inner lobe (4), outer lobe (4), and center lobe (1) according to their relationship to the serpentine fuel annulus. In-pile tubes up to five inches in overall diameter can be accommodated in core section locations other than the center lobe and lobes that contain safety rods.

Twenty inner capsule irradiation positions are located within the core reflector tank. Two outer reflector facilities contain drilled aluminum blocks to handle capsules.

Reactor vessel penetrations provide entry and exit locations for the nine loops, instrumentation lead out, and experiment removal facilities. The top-head closure plate arrangement provides a unique servicing area for the nine loop experiments. Instrumentation and experiment change out is performed in a central area of ready access.

Nine shielded cubicles are provided in the basements for out-of-pile equipment associated with the various experiments. Additional space is provided for experiment control consoles in the basements.

Reactor control circuitry is available to all experiments as required by safety considerations.

2. PRESSURIZED WATER LOOP FACILITIES

2.1 General Description

Six pressurized water cooled test loops occupy the center, three inner, and two outer flux trap positions in the ATR core as shown in Figure IV-3-1. Each in-pile loop is a closed piping system designed for bottom reentrant coolant flow through a pressure tube located within the reactor vessel. Since each loop is designed to operate over a wide range of nuclear, thermal, hydraulic, and chemical conditions, a varied experimental program can be conducted.

Each experimental facility is equipped with an in-pile pressure tube assembly, equipment cubicle, sample area, and control console. The loop safety system is tied in to the reactor control system.

FIG. IV-3-1 IN-PILE LOOP ARRANGEMENT.

2.2 Pressure Tube Assembly

The loop pressure tube assemblies consist of in-pile and subpile room components joined by welding during installation to form a pressure system which extends from above the reactor vessel, through penetrations in the top and bottom reactor closure plates to the primary coolant piping trench in the subpile room floor. This arrangement is shown in Figure IV-3-2.

The in-pile component of the six pressure tube assemblies is identical. Instrument leadout conduit and mechanical pressure tube closure are located above the reactor top closure plate for access during experiment or instrument change. The pressure tube containing an integral flow baffle tube extends downward through the reactor bottom closure plate.

Primary loop coolant enters the pressure tube assembly via the inlet piping connection in the floor of the subpile room and is directed upward through the flow tube and specimens. Below the reactor top closure plate, the coolant passes through ports provided in the flow tube, reversing direction, flows downward inside the annulus formed by pressure tube and flow tube to the outlet piping connection in the subpile room floor.

An external insulating jacket surrounds the pressure tube from its juncture with the top closure housing through the bottom closure plate and terminates at a

FIG. IV-3-2 PRESSURIZED WATER IN-PILE TUBE ARRANGEMENT.

packing gland seal formed with the pressure tube below the reactor bottom closure plate. This seal is designed to accommodate differential vertical expansion between the pressure tube and insulation can. The annular space between the jacket and pressure tube contains helium as an insulating medium. Inlet and outlet connections are provided in the packing gland seal fitting. A small flow of helium is metered to each of the six in-pile tube assembly annuli continuously and discharged to individual loop moisture monitors and finally exhausted to the reactor stack. This system detects jacket or pressure tube leaks. The insulation jacket is protected against excessive internal pressure by rupture discs on the outlet line of the helium system.

The subpile room component assembly for each pressure tube location is separate and distinct; none are interchangeable because of the piping trench locations in the subpile room floor. This component assembly consists of pressure tube and flow tube extensions leading from the in-pile components via an angled off-set to the primary piping connections in the subpile room floor. A particle trap arrangement is provided at the bottom of the subpile room pressure tube and is located within a shielded container set below floor level at the end of the piping trench. The subpile room pressure tube is insulated and provided with removable lead shielding from its juncture with the in-pile assembly to the subpile room floor.

Monitoring instrumentation for the in-pile pressure tube consists of recording specimen temperatures (recorders are provided on the instrument panels for 24 specimen temperatures with space for three additional 2-pen recorders), the inlet and outlet primary coolant temperature, the inlet pressure, and the differential temperature and pressure across the specimen. The total system heat load is computed from loop primary system flow rates and the differential temperature measurements.

The pressure tube is supported at its penetration through the reactor top closure plate, being fixed both laterally and vertically at that location. An alignment and locating tie system furnishes lateral support for the tubes at the top of the neck shim rod housing. Guide fins on the insulating jacket inside the snubber housing of the safety rod drives and a packing gland in the bottom closure plate furnish lateral support for the assembly below the core.

The lower end of the offset in the subpile room pressure tube is utilized as an auxiliary vertical support point. A constant load hanger of 475 pounds at this point compensates for the dead weight loads produced by the lead shielding.

The top closure assembly consists of the closure housing, closure sleeve with threaded locking ring, and closure plug with threaded locking ring. The closure housing is an enlarged cylindrical extension of the top of the in-pile pressure tube and extends through the reactor top closure plate.

The closure sleeve is a concentric spacer sealed to the closure housing with dual rubber O-rings and locked in place by a follower ring threaded into the closure housing. Its prime function is to permit use of a full-opening closure plug for future experiments and to provide a renewable seal surface for the closure plug O-rings.

The closure plug closes the top opening of the pressure tube. It is sealed to the closure sleeve with dual rubber O-rings and locked in place by a follower ring threaded into the closure sleeve. The closure plug is keyed to the closure

sleeve which is in turn keyed to the closure housing to prevent rotation and provide a preferred orientation of components. The closure plug is bored to pass the specimen thermocouples and is fitted at its top with a conax-type seal for exit of the specimen thermocouple.

A lifting sleeve attached to the top of the closure plug by the conax seal nut provides a conduit for three feet of excess thermocouple length; an amphenol connector receptable is mounted near its top end for connecting thermocouple leads. The lifting sleeve is vented to atmosphere to allow detection of conax seal leakage and prevent pressure buildup. A lifting adaptor for specimen transfer is permanently attached to the top of the lifting sleeve.

The flow tube in the subpile room component assembly is an integral part of the assembly. It is welded to the pressure tube at the bottom between the inlet and outlet nozzle connections and is positioned through and above the angled offset by radial spacers. The top end of the subpile room flow tube is a socket receiver for the piston rings which seal between the in-pile flow tube and the subpile room portion.

The in-pile assembly flow tube is a removable item and extends from the top closure to the piston ring seal connection with the subpile room flow tube. The piston rings are mounted in external grooves on the bottom of the in-pile flow tube. The top of the in-pile flow tube engages a seat and keyway provided in the lower portion of the closure housing. Holddown is normally provided by the closure sleeves; however, threads are provided in the closure housing for a flow tube locking ring to be used with a full diameter one-piece closure plug. External radial spacers located at approximately three-foot intervals locate the in-pile flow tube within the pressure tube. Threads are provided at the top of the in-pile flow tube for engagement with a lifting tool.

The test assembly, normally consisting of specimens, specimen holders, retainer tube, hanger rod assembly, thermal barrier, and specimen thermocouples, will be provided by the experimenter. The hanger rod assembly with integral thermal barrier and attached specimen train is bolted to the bottom face of the closure plug. The specimen thermocouples thread through the closure plug and conax seal and connect to the amphenol connector receptable provided.

The in-pile pressure tube and jacket of all loops are constructed of type 348 stainless steel. The pressure-containing parts of the in-pile portion of the loop have been designed in accordance with Section VIII, Unfired Pressure Vessels, of The ASME Boiler and Pressure Vessel Code. In addition to the basic code design, the thermal stresses in the in-core portion of the pressure tube assembly produced by gamma heating within the tube walls have been evaluated in accordance with procedures given in PB-151987, "Tentative Structural Design Basis For Reactor Pressure Vessels and Directly Associated Components". Table IV-3-1 lists the conditions used in the design and analysis of the pressure tube. Table IV-3-2 is a summary of important in-pile pressure tube assembly dimensions.

2.3 Out-of-Pile Facilities

2.31 General. The out-of-pile equipment includes the interconnecting piping in the subpile room and pipe corridor and the loop equipment installed in individually shielded cubicles and service areas. The out-of-pile systems are basically the same as the J-10 ETR Loop which has operated successfully since 1959.

TABLE IV-3-1

PRESSURE TUBE DESIGN CONDITIONS

Design Pressure	2500 psig
Design Metal Temperature (basic)	650°F
Uniform Internal Heat Generation (maximum)	32.5 watts/gram
Cycles for Design Pressure Loading	400/lifetime
Cycles for Design Thermal Loading Top of Design Pressure	4200/lifetime

TABLE IV-3-2

IN-PILE PRESSURE
TUBE ASSEMBLY DIMENSIONS

Active Lattice Area	OD (inches)	ID (inches)
Insulation Jacket	2.874	2.634
Pressure Tube	2.529	2.124
Flow Tube	1.875	1.745

The interconnecting piping is welded to the in-pile tube assemblies in the subpile room and leaves the subpile room in shielded trenches in the floor. The piping enters the cubicle through a shielded pipe corridor which surrounds the reactor.

A shielded cubicle with over 500 square feet of floor space and an enclosed sample area with over 300 square feet of floor space have been allotted for each loop on the second basement of the ATR building as shown on Figure IV-3-3.

The inner wall of each cubicle is common to the pipe corridor. The cubicle contains the experimental loop primary heat exchangers, primary circulation pumps, pressurizer, primary loop heaters, purification cooler, regenerative heat exchanger, purification ion exchange columns, pressurizer intercooler, and associated piping and valves. Equipment in the cubicle is so arranged that any major component can be installed or removed without removing adjacent equipment. Each cubicle has an overhead monorail for handling primary pumps and other heavy equipment. The walls of the cubicle are constructed of 32-inch-thick, high density concrete block to reduce radiation levels outside the cubicles to acceptable levels. Each cubicle and sample room is ventilated through an individually controlled supplementary exhaust system.

For three of the loops, the primary out-of-pile process equipment is fabricated from Type 348 stainless steel. For the other two loops the principal out-of-pile process equipment is fabricated from Inconel.

All critical valves such as isolation valves and loop drain valves are located such that they can be operated from outside the cubicle walls by use of extension handles. A sampling box is located in an enclosed sampling room adjacent to each cubicle where sampling connections from the loop are provided.

The loop control panels, electrical switchgear, and auxiliary equipment areas are located on the same floor but outside the respective cubicles. The design permits complete remote operation of all loop equipment.

NOTE:
FOR ILLUSTRATIVE PURPOSES
CUBICLE 2B AND ITS SAMPLE AREA
ARE SHOWN.

FIG. IV-3-3 ATR SHIELDED CUBICLE AND ENCLOSED SAMPLE AREA.

A full scale mock-up of all equipment, piping, and valves within a cubicle has been constructed to demonstrate acceptable methods of remote operation, handling, and maintenance.

2.32 Flow Path. Figure IV-3-4 shows the various flowpaths and components of the loop primary coolant system and the associated instrumentation and valves.

Coolant enters and leaves the in-pile tube via the shielded trench in the subpile room floor. The loop coolant piping is arranged with an inverted U-bend in the pipe corridor between the reactor bottom head and the cubicle. This U-bend serves the purpose of keeping the in-pile section of the pressure tube full of water in the event a leak develops in the cubicle piping. Coolant from the in-pile tube flows through a flow control valve then to a pneumatically actuated three-way valve which proportions flow through and around the primary heat exchangers to maintain a constant inlet temperature to the in-pile tube. From the primary heat exchangers, coolant flows to the suction of the primary circulation pumps, through piping with clamp-on loop heaters, and back to the in-pile tube. The pressurizer is connected to the primary coolant system loop piping at the suction of the primary circulation pumps.

2.33 Electrical. The electrical loads are supplied as follows:

Each primary circulation pump is connected to permit operator selection of commercial or diesel power. Only one pump per loop is permitted to be connected to diesel power at any particular time. Pump control power comes from the same source which provides pump power.

The instrumentation system power supply is designed to provide automatic relay switching between commercial and diesel power.

All other electrical loads are supplied from commercial power with no provision for switching to other supplies.

2.34 Loop Components. All parts of the loop which contain primary coolant have welded joints except that flanged couplings are used for the rupture discs, the flow control valve, and the primary circulation pumps where the pump case joins the motor. The loop strainer is provided with a "Conoseal" type joint for screen removal purposes. The inlet and outlet connections to the ion exchange column contain O-ring seals for strainer removal. The flanges of the pumps are designed to permit seal welding. All welds are made and inspected in accordance with standard procedures.

All primary system valves are rated at 2500 psig and 650°F with zero leakage. They are designed to withstand 2000 thermal transient cycles.

The primary piping layout is such that dead legs and pockets which trap corrosion products are held to a minimum. The primary piping is located at least 18 inches above the floor to simplify loop draining and to facilitate the decontamination of the floors. Such equipment as instrument lines, instrument panels, drain trough, some radiation monitors, and some of the extension handles from valves are in a sample room outside and adjacent to the cubicle.

FIG. IV-3-4 LOOP FACILITIES FLOW DIAGRAM.

The loop pressurizer is provided to accommodate thermal expansion and contraction of the primary coolant, to maintain the operating pressure, and to degas the loop. The pressurizer heaters are designed to raise the temperature in the pressurizer from 70 to 650°F in four hours. A "Heliflow" type intercooler is installed between the loop and the pressurizer to reduce heater requirements during degassing and to minimize vessel thermal shock during loop depressurization.

Each loop contains three centrifugal canned rotor pumps which are installed in series. Normally, two pumps are required and one is an operating spare. Each pump is capable of pumping 60 gpm at 335-foot TDH.

The three primary heat exchangers per loop are cooled by 240 psig, 130°F water (HDW) on the shell side. "Heliflow" type exchangers, designed to withstand thermal shock are used in all loops. Each bank of three is capable of removing approximately 0.9 MW of heat at a primary coolant inlet temperature of 200°F and a flow of 120 gpm.

The primary loop heaters have sufficient heating capacity to maintain the loop, with zero nuclear heating, at an operating temperature of 600°F over the normal range of coolant flows and to raise the coolant temperature from 70 to 600°F in three hours.

The purification system consists of two mixed bed ion exchange columns in parallel, a regenerative heat exchanger, and a purification cooler. Each ion exchanger column is shielded by nine inches of lead shot. Removal of spent resin is accomplished by flushing to a shielded cask.

2.35 Auxiliary Systems. The secondary side of the primary heat exchangers, the purification cooler, and the motors of the primary circulating pumps are cooled by a recirculating treated water system (HDW). In the event of loss of HDW flow to the primary circulation pump motors, all but one pump will stop after a three minute delay. The remaining pump will be run to destruction if required, to maintain primary coolant flow as long as possible.

The loop will be filled initially from the ATR low-pressure demineralized water (LDW) system. During operation, the makeup water is further purified by passing through deoxygenerators and mixed bed demineralizers. Two positive displacement-type makeup pumps are provided for each loop. Two fill pumps, a recirculation pump, and a makeup water storage tank are common to all loops.

The drain systems are valved so that a loop may be drained to either the ATR warm experimental waste (WED) or the hot experimental waste (HED) system. The ATR warm waste system will handle the loop drains when the radioactivity concentrations are low. The hot experimental waste system will be utilized for highly radioactive water.

All primary loop pressure relief devices and the pressurizer vent to the ATR collection system. All primary vent lines extend outside the cubicle to the drain collection trough in the sample area where vent valves are installed. All secondary pressure relief devices vent to the warm drain tank via four- and six-inch lines. HDW system is protected from overpressure by primary coolant in heat exchangers by relief devices in the cubicles.

One common cart-mounted decontamination system, consisting of solution tanks, fill pumps, electric heaters, and agitators, is provided for the water loops.

The system utilizes the feed and bleed principle with solutions of alkaline permanganate and ammonium citrate as decontaminants.

2.36 Reactor Control and Safety System. Each loop variable which can affect the safety of personnel or equipment integrity has an independent instrument system which monitors and continuously records the process variable with provisions to initiate an alarm and a reactor power reduction. A reactor power reduction will be initiated by any of the following conditions:

- (1) Low primary loop flow to in-pile tube.
- (2) Low primary loop flow from in-pile tube.
- (3) Low pressure in pressurizer.
- (4) Low inlet pressure to in-pile tube.
- (5) High differential temperature across in-pile tube.
- (6) High temperature at in-pile tube outlet.
- (7) High temperature at in-pile tube inlet.
- (8) High Specimen temperature.
- (9) Loss of Power Reduction Voltage.
- (10) Diesel or commercial power loss to primary loop circulation pumps.

The annunciation and power-reduction setpoints of the above listed variables, except for instrument voltage, can be changed or tested, either during reactor operation or during shutdown. Provision is made to disconnect the instrumentation for each variable from its power reduction function during setpoint adjustment or testing. With the exception of the high differential temperature across the in-pile pressure tube, the variables are grouped so that when any one is disconnected for setpoint adjustment purposes, a related variable remains in service. If two related variables are inadvertently disconnected at the same time, a reactor power reduction will be initiated automatically. For each of the quantities (1) through (8), the type of power reduction to be initiated (scram, set back, etc) can be selected individually.

2.37 Handling of Loop Test Assemblies. The in-pile test assemblies will be inserted or removed from the pressure tubes only when the reactor is shut down. A water-filled and shielded handling cask will be utilized for transferring irradiated test assemblies from the reactor to the working canal or for transport from the reactor building. Figure IV-3-5 illustrates the general use of the transfer cask and associated auxiliary handling equipment during transfer to the working canal.

A water-filled removal sleeve is provided between the bottom of the cask and the top of the pressure tube to ensure a wet transfer as the assembly is drawn up into the cask. The top-head shield assembly, loop transfer shield plate, and shield cylinder provided over the reactor top head protect personnel during withdrawal of a test assembly from a loop pressure tube into the cask.

FIG. IV-3-5 LOOP HANDLING ARRANGEMENT.

A shielded cask transfer station is provided in the working canal which supports the cask during discharge of the test assembly into the canal. Water-filled test assembly storage cans are provided which seal to the bottom of the cask to ensure a wet transfer as the assembly is discharged into the canal. After discharge of the assembly into the storage can, the can and assembly are transferred out of the transfer station to a storage rack provided along the canal wall.

3. GAS-COOLED LOOP FACILITY

3.1 General Description

A gas-cooled materials and fuel test loop is scheduled to occupy an outer lobe position as shown in Figure IV-3-1. The loop will be used to obtain irradiation data in environmental conditions applicable to gas cooled reactor technology. The following description is based on Title 1 design.

The loop consists of a helium recirculating system to provide high temperature gas flow past experiments located in the active core section and housed in a pressure tube assembly. Special arrangements are provided to isolate the high temperature gas from the pressure tube. Out-of-pile equipment is located in a shielded cubicle. A gas purification system, to reduce coolant impurities to low levels, and loop instrument and control system are provided. Loop safety circuits are connected to the reactor control system.

3.2 In-Pile Tube Assembly

The in-pile pressure tube, fabricated of type 348 stainless steel, extends from the reactor bottom-head closure plate through the top-head closure plate with a branch outlet pipe connection which passes through a reactor side outlet nozzle. Figure IV-3-6 shows the arrangement of the in-pile assembly. The tube is fabricated in four sections and held together by means of Grayloc mechanical joints. The lower tube section contains a Haynes-25 flow tube maintaining a stagnant helium thermal insulation gap between the high temperature flow tube and low temperature pressure tube. A teardrop plug located in the flow tube below the core is used for shielding. The core section of the flow tube is fabricated of columbium because of its superior high temperature properties. This section is removable since radiation damage effects are not completely understood. The upper section of the pressure tube is larger in diameter than the two lower sections to allow for removal of the columbium flow tube section, the lower shield plug, and Haynes-25 flow tube. The flow tube section above the core is constructed of Hastelloy X and provides an annulus for attemperation flow between itself and the increased diameter pressure tube. The upper section of the in-pile tube contains the specimen hanger rod, thermal shields, and neutron and gamma shielding to minimize radiation streaming at the reactor top head.

High temperature helium gas enters the in-pile tube below the reactor bottom-head closure plate and passes upward inside the various flow tubes arriving at a test section at a temperature of about 1800°F. A materials or fuel test specimen with a maximum output of 300 kW increases the helium temperature to 2000°F at the outlet of the core section. Attemperation helium, introduced at the side arm reactor nozzle outlet, flows counter to the main gas stream in the

FIG. IV-3-6 GAS LOOP IN-PILE TUBE ARRANGEMENT.

annulus between pressure tube and flow tube and joins the main gas stream above the core section. The combined gas stream passes up and finally out of the reactor through the side arm nozzle at a temperature of 1600°F or less.

3.3 Out-of-Pile Equipment

Most of the primary system equipment is located in a shielded cubicle in the reactor basement. Necessary utilities such as air, demineralized water, and power are available.

Six gas circulators, piped in series, pump helium at a rate of 5000 lbs/min at 80 percent of capacity. Helium gas at 175 psig and 200°F inlet conditions is discharged at 255 psig and 320°F. The 85 hp centrifugal circulators have gas cushion bearings. The gas serves as coolant and lubricant eliminating the necessity of oil or grease that could contaminate the gas stream. The motor and impeller are mounted on a common hollow shaft permitting gas recirculation flow to cool the unit while in operation. Water coils around the sealed casing cools the recirculation gas. Maximum circulator speed is 15000 rpm with normal operation at 12000 rpm. Failure of one circulator automatically speeds up other units to maintain constant gas flow rate. Speed control is accomplished by means of frequency regulation.

Six electric heaters supply a maximum of 175 kW each to the gas stream. The heaters are capable of increasing the gas temperature to a maximum of 2000°F. The heating coils consist of molybdenum and Nichrome elements and castable fibrefrax electrode insulation. The heaters are enclosed in horizontal casings consisting of an inner liner of Haynes 25 alloy and an outer liner of 316 stainless steel and Hastelloy X.

A regenerative heat exchanger system provides the means to transfer heat from the hot gas stream leaving the reactor in-pile tube assembly to the cool recirculated gas stream before entering the electric heater. This system reduces the loop heat load caused by the low gas temperatures required in the recirculators.

A primary heat exchanger further cools the gas stream leaving the regenerative heat exchangers to 200°F before entering the circulators.

A shielded 2 micron filter, designed for high temperature, handles the gas stream leaving the in-pile section. A second low temperature filter is located between the primary heat exchanger and the suction to the first circulator. The discharge helium line from the circulators is divided to provide attemperation (cooling) gas to the side arm flow tube-pressure tube annulus.

High temperature (2000°F) helium leaving the electric heaters is channeled to a flow tube of Haynes-25 alloy housed inside type 316 stainless steel pressure tube piping. This assembly mates with the in-pile tube assembly. The annulus formed between the flow tubes and pressure tubes of both in-pile and out-of-pile assemblies is pressurized by means of a shroud gas system. This system senses differential pressure between the main helium stream and the annulus. This signal controls a bleed and feed arrangement composed of helium cylinders, control valve, controller, and vent system. This arrangement reduces the mechanical strength requirements for the flow tubes operating at high temperature.

The primary gas impurities concentration is limited to one vpm by passing a side stream of the primary gas flow through one of two identical purification systems.

System pressure is maintained by feeding helium from a makeup system or bleeding helium to an exhaust tank.

A holdup system, consisting of shielded tank and compressor, is provided to retain contaminated loop helium inventory.

A flow sheet of the primary helium system is shown in Figure IV-3-7.

Helium at 1600°F maximum temperature leaving the reactor side arm nozzle passes through the shielded filter into the regenerative heat exchanger system. The gas flow, reduced in temperature, is further cooled in the primary heat exchanger. After passing through the second filter, the gas stream enters the suction of the circulators.

The gas stream leaving the circulators is divided with attemperation gas flowing to the reactor discharge nozzle and the main stream entering the regenerative heat exchanger system. The heated gas is fed to the electric heaters and finally enters the in-pile section. A small side stream, taken from the circulator discharge line, flows through the purification system and is returned to the main stream at the circulator suction line.

3.4 Instrumentation and Reactor Control

Loop control is accomplished with conventional instruments and control elements to provide a wide operating range of temperature, pressure, and flow. Loop and experiment protection is accomplished by design and reactor control circuitry. This circuitry connects loop parameters to reactor power reduction systems. Each experimental parameter is analyzed and set points adjusted to reduce reactor power before hazardous conditions occur.

The gas cooled loop design criteria are:

	<u>Materials</u>	<u>Fuels</u>
Coolant Gas -- lb/hr max.	4140	4140
Attemperation -- lb/hr max.	800	800
Heat Generation -- Gamma -- kW	300	150
Fission -- kW		150
Test Specimen -- Length -- inch	48	48
Specimen Inlet Temperature -- °F (Max.)	1800	1800
Specimen Outlet Temperature -- °F (Max.)	2000	2000
Total Loop Pressure Drop -- psi	80	80

4. INNER AND OUTER REFLECTOR CAPSULE FACILITIES

4.1 Inner Reflector Capsule Facilities

The twenty inner reflector capsule facilities consist of four 5-inch diameter and sixteen 3.25-inch diameter holes penetrating the beryllium reflector the entire core length. Beryllium filler pieces are provided for the facilities when not in use.

4.2 The Outer Reflector Irradiation Facilities

Two outer reflector irradiation facilities located 180 degrees apart in a north-south direction outside the core reflector tank accommodate 12 and 22 aluminum blocks, respectively. The aluminum blocks are drilled to provide five 3-inch, one hundred and twenty-nine 1.312-inch, and four 0.875-inch diameter positions.

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V. REACTOR CORE

V-1 Core Description

V-2 Nuclear Characteristics

V-3 Thermal and Hydraulic Characteristics

V-4 References

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V-1. CORE DESCRIPTION

1. GENERAL CORE CONFIGURATION AND SUPPORT

1.1 Description of Components and Method of Support

The ATR core region contains a serpentine arrangement of 40 fuel elements which provides an annulus of uniform width between and around nine flux trap areas. The flux trap areas are located in a 3-x-3 square array. The active height of the core is four feet. The arrangement plan is shown in Figure V-1-1 and schematically in Figure V-1-2.

FIG. V-1-1 CORE ARRANGEMENT PLAN.

FIG. V-1-2 SCHEMATIC CROSS SECTION AT CORE MIDPLANE.

The elements are bottom supported by a slotted flange on the lower end of the adjacent flux trap baffle. Each slot receives a T-bar support key on the fuel element lower end fitting for accurate bottom positioning. The upper end of each fuel element is positioned in the same manner by a slot in the flux trap baffle which accepts an extension on the top end fitting of the fuel elements.

The flux trap baffles are supported and oriented by tubular extensions on the flow distribution tank. This arrangement provides orientation and location of the inner boundary of the fuel annulus at the bottom of the core and establishes vertical references. The four inner flux trap baffles are fixed to the neck shim housing at the top of the core by means of dovetail keys. Since the neck shim housing and inner flux trap baffles are located by a common base, this portion of the core structure is semirigid. The top ends of the outer flux trap baffles are positioned and supported laterally by a connection with the outer cylinder gear box support beam. The center rib of the fuel element top end fitting acts as a spacer between the gear box support beam and the inner flux trap baffles and between the outer flux trap baffles and the neck shim housing at the top of the core. This spacer arrangement is used to prevent compressive loading on fuel element side plates below the top end fitting that could result from 100 percent stack-up of manufacturing and assembly tolerances. The top of the fuel element annulus is, therefore, controlled within specified limits. The center flux trap baffle is located inside the neck shim housing.

The outer radius of the 24 fuel elements, supported by the four inner flux trap baffles, is adjacent to the inner face of the beryllium reflector. The beryllium reflector surrounds the core area and consists of eight segmented blocks 51 inches high. These blocks form a cylinder approximately 51 inches in diameter, and are bottom supported on a ledge at the bottom of the core reflector tank. The reflector inner contour follows the fuel annulus around the inner flux traps

The four flux trap positions on the corners of the square, enclosed by the fuel annulus, are referred to as inner flux trap positions, whereas the four side positions located outside the fuel annulus are referred to as outer flux traps. The central position is referred to as the center flux trap.

The fuel annulus is bounded by eight flux trap baffles, the neck shim housing, and the inner face of the beryllium reflector blocks. The center flux trap baffle is located in the center of the neck shim housing.

The 40 fuel elements are each constructed as a 45-degree annular sector of a right circular, hollow cylinder with inner radius equal to the outer radius of the flux trap baffles. The

and the periphery of the outer flux trap baffles. Each of the eight blocks contains two vertical, cylindrical cavities to accommodate the rotating beryllium-poison control cylinders. The top of the beryllium blocks and cylinders is located by the gear box support beams on the top of the core reflector tank.

The fuel annulus clearance has been designed to accept fuel elements with maximum fabrication tolerances applied to the fuel elements, beryllium reflector blocks, and flux trap baffles. The fuel locations with respect to experiments are retained irrespective of fuel annulus deviations caused by the fuel-baffle support arrangement.

Coolant flow through the ATR core is downward whenever the primary or emergency pumps are operating. The pressure drop through the core, therefore, acts to retain the elements in position. Certain emergency conditions can initiate a maximum of 2000 gpm coolant flow upward through the core. The pressure differential developed by this flow is less than 0.4 psi. The maximum resulting upward force on a fuel element (9-square-inch cross sectional area) is 3.6 pounds. Since a fuel element weighs 15 pounds in water, a safety factor greater than four exists to prevent lifting.

1.2 Tolerance Analysis

An analysis of critical component tolerances was made based on design dimensions.

(1) Reflector outer radius -- Reflector tank inside diameter

Nominal gap	-- 0.062 inch
Minimum gap (limited by pads on Tank ID)	-- 0.022 inch
Maximum gap	-- 0.101 inch

(2) Beryllium-Fuel Annulus Surface Variations

At top of core (with reference to core reflector tank assembly)

Toward the fuel -- 0.025 inch

Away from the fuel -- 0.029 inch

At bottom of core

Toward fuel -- 0.022 inch

Away from fuel -- 0.026 inch

(3) Neck Shim Housing -- Fuel Annulus Surface Variations

At top of core (with reference to core reflector tank assembly)

Fuel element upper end fitting acting as a spacer

Toward fuel -- 0.039 inch (Flux trap baffle as far
 toward tank as possible)

(with reference to outer flux trap baffle)

-- 0.021 inch (Flux trap baffle in
 nominal position)

Away from fuel -- 0.061 inch

(4) Inner Flux Trap Baffle Outside Diameter Variations

- At bottom of the core (with reference to flow distribution tank)

Toward the fuel -- 0.011 inch

Away from the fuel -- 0.015 inch

At the top of the core (with reference to the core reflector tank)

Fuel element upper end fitting acting as a spacer between inner
trap baffles and beams

Toward the fuel -- 0.035 inch

Away from the fuel -- 0.039 inch

(5) Outer Flux Trap Baffle Outside Diameter Variations

At the bottom of the core

Toward the fuel -- 0.011 inch

Away from the fuel -- 0.015 inch

At the top of the core

Toward the fuel -- 0.012 inch

Away from the fuel -- 0.018 inch

(6) Clearances -- Flux Trap Inside Diameter to Safety Rod Guide
Ribs

(smallest baffle inside diameter)

Nominal diametral
clearance -- 0.125 inch

Minimum diametral
clearance -- 0.120 inch

Maximum diametral
clearance -- 0.135 inch

(The above clearances do not apply to the full-up position which is controlled by ribs on the experiment insulating can. Minimum radial clearance in the full-up position is 0.042 inch.)

(7) Safety Rod ID -- Experiment Insulation Can OD
Clearances

Nominal Radial
Clearance -- 0.250 inch

Variations in Safety
Rod -- 0.052 inch

Variations in Insulation
Can -- 0.037 inch

Minimum Radial
Clearance -- 0.161 inch

2. FUEL ELEMENTS

The ATR fuel arrangement geometry in the core requires a circular sector fuel element. Each element consists of 19 parallel curved fuel plates attached to the two side plates to form a 45-degree sector of a right circular cylinder. The fuel element is shown in Figures V-1-3 and -1-4. Forty elements are required for a complete core loading.

Each of the 19 fuel plates is 49.5 inches long with an active fuel length of 48 inches. The 17 inner fuel plates (2 through 18) are 0.050 inch thick with 0.020 inch of fuel and 0.015 inch of cladding. The nominal coolant channel width between plates is 0.078 inch. The two outermost fuel plates (1 and 19) of each element are designed with extra thick clad to withstand the greater pressure differential resulting from variability of the coolant channels formed by the outer fuel plates, beryllium reflector, flux trap baffle, and neck shim housing. The outside plate with the greatest radius of curvature (plate 19) is 0.100 inch thick with 0.020 inch of fuel and 0.040 inch of clad. The inside plate with the smallest radius of curvature (plate 1) is 0.080 inch thick with 0.020 inch of fuel and 0.030 inch of clad.

The side plates are vented by localized slots to provide interconnection between all channels and thereby reduce lateral pressure differentials. A typical arrangement is shown in Figure V-1-3. Pads are included on the exterior surface of the side plates to ensure adequate coolant channels between elements and provide a minimum flow path for interconnecting flow through the vents and external channels.

The fuel element end fittings are designed for support and positioning on the flux trap baffles. The end fittings are attached to the fuel element side plates by welding and provide minimum restriction to coolant flow through the element. In addition to providing a lifting handle, the top-end fitting includes a curved baffle to direct coolant flow into the fuel channels with a minimum of turbulence and to prevent mal-distribution. The top-end fitting also acts as a spacer

- ① T-BAR SUPPORT KEY
- ② UPPER END FITTING
- ③ LOWER END FITTING

- ④ SIDE PLATE VENTS
- ⑤ SIDE PLATES (ALUMINUM)
- ⑥ FUEL PLATES (ALUMINUM CLAD)

TOP VIEW

SECTION Z-Z

BOTTOM VIEW

FIG. V-1-3 FUEL ELEMENT DETAILS.

FIG. V-1-4 FUEL ASSEMBLY AND DETAILS

to position the top end of adjacent flux trap baffles in lieu of an upper grid plate. A comb is positioned across the top and bottom mid span edges of the fuel plates to provide additional support.

The fuel is a dispersion of 41 weight percent U_3O_8 in an X8001 aluminum matrix. The fuel matrix contains approximately 0.19 weight percent B_4C as a burnable poison. The fuel matrix is clad with an aluminum alloy using conventional picture frame techniques.

2.1 Materials

Table V-1-1 lists the fuel element component materials.

The fully annealed 6061 aluminum clad condition was chosen because of possible overaging effects during irradiation.

TABLE V-1-1

FUEL ELEMENT MATERIALS

Fuel Matrix	U_3O_8 dispersed in X-8001 aluminum
Fuel Matrix -- Clad Bonding Agent	1100 aluminum
Cladding	6061 -- O aluminum
Burnable Poison (in matrix)	B_4C
Side Plates	6061 -- T6 aluminum
End Fittings	356 aluminum

2.2 Description of ATR Fuel Element Fabrication Techniques

The general fabrication process for ATR fuel elements is:

- (1) Intermix U_3O_8 fuel, the B_4C burnable poison, and the aluminum (X8001) powder by dry blending
- (2) Press the powders to produce a fueled compact

- (3) Vacuum anneal the compact
- (4) Assemble the compacts into a billet using the one-piece, picture frame technique
- (5) Hot roll-bond the billet to form a fuel plate
- (6) Cold roll to final thickness
- (7) Anneal the plates
- (8) Form the plates to the specific radius of curvature
- (9) Assemble the fuel plates to the side plates by roll swaging, add end fittings, and machine as required.

Some details of the fabrication techniques for the ATR fuel element are presented.

A pressure of 45 tons/in² is required for pressing compacts. Maximum compact thickness is 0.365 inch.

Individual weighing of powder for each plate leads to improved fuel tolerance control. The aluminum powder is vacuum degassed and screened prior to blending to control aluminum segregation in the compact. The compacts are vacuum annealed at 590°C for three hours to drive off impurities which could cause blisters to form within the fuel during subsequent annealing. As-pressed density was controlled to 92.5 ± 0.5 percent of theoretical.

Dimensional control of the uniformity of fuel compact thickness to ± 0.0015 inch is required to control the shape of the fueled area of the finished plate.

The bondability of type 6061 aluminum was improved by cladding the mating faces and picture frames with a five percent thickness of 1100 aluminum.

Reduction ratios during the rolling operations are limited to ten to prevent fragmentation and stringering of fuel particles.

Radiography was used to inspect the fuel plate for dimensions of the fuel area and to locate the fuel for final machining of the edges.

Based on the results of the fabrication development, fuel plates are made to specification with a process yield of 95 percent. Included in this is a reject rate of less than 0.5 percent. Blister detection was performed using ultrasonic equipment capable of detecting nonbonds 1/16 inch and larger in diameter.

Forming of plate curvatures is accomplished using two passes on a mar-forming die with the plates in the annealed condition. A relationship for spring-back was established to control deviations in chord height to less than 0.008 inch.

Grooves in the side plates were made with a milling cutter which gang mills the slots with sloping sides to match the varying angles of the fuel plate edges.

The plates, after degreasing and bright dipping in acid, are joined to the side plates by roll swaging. During the swaging operation, spring loaded rollers are used to hold the center of the fuel plate down against temporary spacers as an aid in controlling coolant channel tolerances. Results of measuring coolant channel deviations of a typical element are shown in Table V-1-2.

TABLE V-1-2

COOLANT CHANNEL DEVIATIONS

Channel Between Plates	Plane A		Plane B		Plane C	
	Average Spacing (mils)	Maximum Deviation from Average Spacing (mils)	Average Spacing (mils)	Maximum Deviation (mils)	Average Spacing (mils)	Maximum Deviation (mils)
1-2	81	+3 -6	81	+3 -4	81	+1 -2
2-3	77	+2 -2	80	+2 -4	78	+1 -1
3-4	80	+2 -2	78	+2 -1	79	+2 -1
4-5	78	+2 -1	77	+2 -1	79	+2 -1
5-6	78	+2 -0	79	+2 -2	77	+2 -0
6-7	80	+2 -2	82	+2 -4	80	+2 -2
7-8	79	+1 -1	76	+1 -2	79	+2 -1
8-9	77	+1 -1	78	+3 -2	77	+0 -1
9-10	80	+2 -2	80	+2 -2	81	+2 -2
10-11	78	+1 -1	79	+1 -1	77	+1 -2
11-12	76	+2 -1	80	+1 -2	77	+1 -2
12-13	82	+0 -2	78	+3 -3	80	+1 -1
13-14	78	+1 -1	78	+1 -1	78	+2 -0
14-15	76	+1 -2	78	+2 -2	75	+1 -1
15-16	81	+3 -1	77	+1 -1	81	+2 -2
16-17	78	+3 -1	78	+2 -2	79	+1 -2
17-18	76	+2 -1	79	+2 -3	76	+1 -1
18-19	78	+1 -3	77	+2 -2	78	+1 -2

Planes A and C are along the axial length 1/2 inch from each side plate. Plane B is along the midspan of the plates in the axial direction. At least 24 equally spaced measurements were made along each plane.

In addition to roll swaging of fuel plates to side plates, joining by welding has also produced satisfactory fuel elements.

Procurement of satisfactory ATR fuel elements from commercial sources by fabrication techniques described has been accomplished.

2.3 Fuel Element Hydraulic Development

An extensive hydraulic development program was performed to investigate ATR fuel element performance and provide data to improve the design. Experiments and analytical work were as follows:

- (1) Pressure deflection tests
- (2) Analytical program covering fuel plate deflection
- (3) Hydraulic buckling tests
- (4) Thermal distortion testing
- (5) Hydraulic proof tests.

2.31 Pressure Deflection Tests -- Static and Dynamic. Pressure deflection experiments included: static single plate tests to confirm the effect of membrane force, static side plate spring rate tests to evaluate membrane force, static full element tests to confirm analytical procedures and evaluate performance, and dynamic hydraulic pressure deflection tests to evaluate hydraulic weakening and confirm static testing methods.

The details of the pressure deflection tests have been reported [1].

The results of the single plate tests confirmed the calculated values based on membrane force.

The spring rate test results to evaluate membrane force were in reasonable agreement with calculated values in the elastic range.

Full element test results showed good agreement with calculated values in the elastic region. The dynamic hydraulic test results were 20 percent higher than the calculated values as shown in Figure V-1-5.

2.32 Analytical Program. The analytical program developed in conjunction with the above tests applied the beams-on-elastic foundation theory. The beams-on-elastic foundation theory assumes that the foundation is elastic and of uniform intensity with a constant foundation modulus. Each plate in the ATR fuel element has a different geometry, and it is, therefore, necessary to assume that an averaged foundation exists across the full width of the side plate. This calculation is performed by determining the importance of each individual plate as a spring relative to the loaded fuel plate. The spring rate is calculated using Castigliano's theorem [2]. The various deflections are then calculated.

The hydraulic deflection tests and above calculations indicated an average deflection rate for plate 19 of 1.26 mils/psi at room temperature. Correcting this value for direction of deflection and temperature yields a value of 1.78 mils/psi pressure differential.

2.33 Hydraulic Buckling Tests. The details of the hydraulic buckling tests have been reported [3].

A summary of the test results is shown in Table V-1-3. The coolant channel outside plate 19 (channel 20) was varied in the tests to simulate ATR conditions.

The results show that the ventilated ATR fuel element is capable of withstanding greater than normal pressure differentials without significant permanent distortion.

FIG. V-1-5 HYDRAULIC PRESSURE-DEFLECTION TEST FOR PLATE NUMBER 19 WITH CONCAVE PRESSURE AT ROOM TEMPERATURE.

Additional ventilated ATR fuel element buckling tests at high temperature (420°F) showed no gross permanent distortion of fuel plates at pressure differentials above normal requirements.

2.34 Thermal Distortion Testing. The temperature gradient across the width of fuel plates can cause deformation or ripple. Simulated ATR fuel plates containing depleted U_3O_8 fuel were subjected to stresses approximating thermal gradient induced stresses expected during operation. The details of these tests have been reported [4].

Results of these tests showed no severe rippling or distortion occurred when the plates were subjected to stresses expected during ATR operation.

TABLE V-1-3

HYDRAULIC BUCKLING TESTS RESULTS

Test Specimen and Channel 20 Description	Overall Pressure Drop ATR Design -- 100 psi (psi)	Maximum Plate 19 Permanent Distortion (mils)
Unventilated initial ATR design, Channel 20 -- 90 mils parallel	100	81
Unventilated ATR design, Channel 20-tapered 75 mils small at top	102	18
Unventilated ATR design, Channel 20-tapered 75 mils small at top	110	22
Unventilated ATR design, Channel 20-tapered 75 mils large at top	110	22
Ventilated ATR design, Channel 20-tapered 90 mils large at top	145	4

2.35 ATR Fuel Element Hydraulic Proof Tests. Hydraulic testing of fuel elements fabricated in production lots was undertaken to assure that the plans and specifications resulting from the fuel element research and development programs would provide fuel elements to withstand the relatively severe operating conditions. This section describes briefly the facility used for this purpose, the experimental apparatus, techniques, and the results to date derived from the test program.

Several, but not necessarily all of the fuel elements used in each reactor core loading, will be tested hydraulically to provide assurance that the fabrication procedures are satisfactory. The first ten elements from each fuel contract will be tested, and, depending upon the performance, testing will be either continued or a statistical sampling procedure instituted. The first ATR core loading will be fully tested.

(a) Test Apparatus. The test facility is a high temperature, high pressure, stainless steel, closed loop circulating demineralized and degassed water. Basic features of the loop include: an electrically driven main loop pump, rated at 1200 gpm at 700 ft TDH; control heat exchanger; cool-down heat exchanger; pressurizer and heaters; and a stainless steel removable test section designed to receive one prototype ATR fuel element. Instrumentation is provided to control and record operating conditions including: test section pressure drop and flow, test section temperature, and primary loop pressure and flow.

Primary loop temperature is controlled from 150 to 500°F by the control heat exchanger. The continuous heat input from the main loop pump is used to initially heat the loop and maintain the primary loop temperature during operation. Cooling water is supplied to the shell side of the exchangers from the site, low pressure, demineralized water system. Test section pressure drops and, thus, flow is controlled by a differential pressure control valve which bypasses the test section. Primary coolant pressure is maintained by the pressurizer which "floats" on the main loop. The upper portion of the pressurizer contains

a steam dome and the lower portion contains primary water. Pressure is increased by activation of two immersion-type electric heaters.

In addition to the main loop instrumentation, the test section is equipped with magnetic pickup transducers located at the midspan of the outer fuel plates and positioned at three axial locations. These transducers provide the data to determine element shift, element bowing, and plate 19 deflection. Mechanical depth probes located at each side plate and at several axial locations provide data to determine element shift, element bowing, and channel 19 thickness. The test section is also designed to admit pitot and static tubes into the fuel element channels to measure interchannel velocities and pressure differentials.

The test section supports the fuel element upper and lower end fittings as in the reactor. With the element installed, channels are formed around the periphery simulating reactor conditions. The inside and outside test section plates can be adjusted to desired thicknesses of channels 1 and 20; the side channels are not readily adjustable.

(b) Test Procedures. The hydraulic test of each fuel element consisted of two parts: (a) the measurement and calculation of interchannel velocity profiles; and (b) a destruction run to determine the structural adequacy of the different types of fuel elements. The same basic test procedure was followed for each element, each test being comprised of the following steps:

(1) Channel widths at several positions were measured for all interior channels at three-inch intervals over the fuel plate length.

(2) The element was installed in the test section and the inside and outside test section plates adjusted until the desired configurations were obtained for channels 1 and 20. Channels 1 and 20 and the side channels were then measured over the fuel plate length at three-inch increments.

(3) The element was instrumented with pitot and static tubes in several channels and installed in the loop. Velocity and pressure measurements were then taken at critical points axially along the element with the loop operating at 150°F and 100 psig across the test section. Upon completion of these measurements, the element was removed from the loop and inspected in the test section. All channels were measured again and the element returned to the loop for the destructive test.

(4) For the destructive test, stationary pitot and static tubes were placed in several channels at locations shown to be critical by the above test. The loop temperature was increased to 350°F and starting at about 500 gpm, the flow was increased in increments of approximately 50 gpm until failure occurred. If the flow capacity of the loop was reached before the element failed, the flow was decreased to the 100 psi value and the temperature increased to 400°F. The flow was then increased in increments to failure. If failure still did not occur, flow was again reduced and the temperature was increased to 425°F.

Deviation of any of the several measurements from expected behavior provided indication of failure.

(5) After indication of failure was observed, the fuel element was removed and carefully inspected for damage. The channels were measured and the data compared with the measurements made before the test. If the element had not failed, it was returned to loop for further testing.

(c) Test Results The results of these tests are shown in Table V-1-4.

Plate deflections at nominal flow conditions were found to be very small and localized. Some shifting and bowing of the entire element was observed at high flow rates. The effect was not important at nominal flow conditions.

The welded element performance showed stronger fuel plate -- side plate joints compared to the roll swaged joints.

The test results indicated the need for certain modifications to improve performance reliability. These modifications include:

- (1) Redesign of the comb attachment
- (2) Improvement in the quality of the roll swaged joint.

Additional testing currently in progress will determine any further modifications that are necessary to ensure the reliability of the ATR fuel element design.

2.4 ATR Fuel Element Stress Analysis

2.41 Introduction. A theoretical analysis of ATR fuel elements was made to determine fuel plate stresses and deflections at reactor operating conditions which produce thermal gradients not present in the isothermal static and hydraulic tests. Azimuthal pressure and thermal stress and vertical plate deflections were calculated for plates 18 and 19 for several channel configurations. Axial thermal stresses were calculated for plates 18 and 19 in two cases: (a) the element with the highest peak power, and (b) the element with the highest peak power including hot spot-hot track factors.

2.42 Pressure-Thermal Azimuthal Stress and Vertical Plate Deflection Analysis. Only plates 18 and 19 were analyzed, since these are potentially the weakest plates in the ATR element. Items calculated included: (a) plate 19 center-span azimuthal pressure stress, (b) plate 19 center-span azimuthal thermal stress, (c) plate 19 vertical pressure deflection, (d) plate 19 vertical thermal deflection, (e) plate 18 center-span total azimuthal stress (due to pressure differential across plate 19 and induced by plate 19), and (f) plate 18 vertical deflection induced by plate 19. Stresses and deflections were calculated by a method developed for the ATR plate geometry and reported in the literature [5]. The method was modified to include a fixed end-movement for plate 19. For the range of differential pressures and temperatures involved, the calculations are good for both roll-swaged

TABLE V-1-4

DESTRUCTIVE PROOF-TESTS OF ATR FUEL ELEMENTS

<u>Fuel Element and Outside Channel Description</u>	<u>Test Conditions (Normal Flow @100 psi -- 675 gpm)</u>	<u>Post Inspection Results</u>
Welded element with final design vent arrangement	1235 gpm @350°F	No evidence of failure
	1265 gpm @400°F	No plate failure, top
	Installed new comb	Comb badly broken
0.190 inch, Channel 20	1275 gpm @425°F	Evidence of failure
0.108 inch, Channel 1	1295 gpm @425°F	Plate 1 broke loose from side plate
Welded element with no venting on side plates	1050 gpm @350°F	Plate 12 buckled toward Plate 11, near top, and toward Plate 13, near center
0.113 inch, Channel 20		
0.108 inch, Channel 1		
Welded element with modified vent arrangement	1185 gpm @350°F	Failure by partial closure of Plate 19
0.190 inch, Channel 20		Failure of upper comb
0.108 inch, Channel 1		
Roll swaged element with modified vent arrangement	1235 gpm @350°F	Evidence of failure
	1257 gpm @350°F	Many plates broken loose
0.063 inch, Channel 20		Comb did not fail
0.108 inch, Channel 1		
Modified comb design		
Roll swaged element with final design vent arrangement	1073 gpm @350°F	Plate 17 buckled and many other plates broke loose from the side plates
0.190 inch, Channel 20		
0.108 inch, Channel 1		
Roll swaged element with final design vent arrangement	1300 gpm @350°F	Some evidence of failure
	1250 gpm @400°F	Plate 17 sheared from side plate
0.175 inch, Channel 20		Upper comb failed
0.108 inch, Channel 1		
Roll swaged element with final design vent	1150 gpm @350°F	Evidence of failure
0.175 inch, Channel 20	1200 gpm @350°F	Plates 15, 16, and 17 pulled out of side plate
0.108 inch, Channel 1		Upper comb cracked
Roll swaged element with final design vent	810 gpm @150°F	Plate 18 pulled out of side plate
0.175 inch, Channel 20		
0.108 inch, Channel 1		
Roll swaged element with final design vent	830 gpm @350°F	Evidence of failure
0.075 inch, Channel 20	900 gpm @350°F	Plates 9 and 17 pulled out of side plate
0.108 inch, Channel 1		
Roll swaged element with final design vent	1150 gpm @350°F	Comb broke and pushed into plates
0.075 inch, Channel 20		
0.108 inch, Channel 1

FIG. V-1-6 PROPERTIES OF 6061 ALUMINUM ALLOY.

and welded-joint elements. Temperatures and pressure differentials were obtained from IBM computer program MACBRE 40.0041, which was written for the ATR fuel element.

Material properties used in the calculations were based on values shown in Figures V-1-6, -7, and -8 obtained from the literature [6, 7, 8].

FIG. V-1-7 PROPERTIES OF 6061 ALUMINUM ALLOY.

FIG. V-1-8 PROPERTIES OF 6061 ALUMINUM ALLOY.

The following assumptions were made:

- (1) Fuel plate material is homogeneous 6061-O Aluminum.
- (2) The elastic limit is not exceeded.
- (3) All joints are completely fixed against rotation, except that the maximum end-moment for plate 19 is equal to 0.4 of the fully plastic moment.
- (4) The side plate temperature is 140°F for all calculations.
- (5) The average temperature of plates 1 to 18 is 350°F for determining the modulus of elasticity E for high peak power element.
- (6) The average temperature of plates 1 to 18 is 475°F for determining E for the hot spot, hot track, peak power element.
- (7) Azimuthal stress at the azimuthal plate centerline is the critical stress.

Calculations were made for the following problems:

PROBLEM 1 Peak power element. Channel 20 -- 0.063 inch, channels 1 to 19 -- 0.078 inch

PROBLEM 2 Hot spot, hot track, peak power element. Channel 20 -- 0.063 inch, channels 1 to 19 -- 0.078 inch

PROBLEM 3 Peak power element. Channel 20 -- 0.113 inch, channels 1 to 19 -- 0.078 inch

PROBLEM 4 Hot spot, hot track, peak power element. Channel 20 -- 0.113 inch, channels 1 to 19 -- 0.078 inch

PROBLEM 5 Peak power element. Channel 20 -- 0.175 inch, channels 1 to 19 -- 0.078 inch

PROBLEM 6 Hot spot, hot track, peak power element. Channel 20 -- 0.175 inch, channels 1 to 19 -- 0.078 inch

PROBLEM 7 Peak power element. Channel 20 -- 0.063 inch, channels 18 and 19 -- 0.072 inch, channels 1 to 17 -- 0.078 inch

PROBLEM 8 Hot spot, hot track, peak power element. Channel 20 -- 0.063 inch, channels 18 and 19 -- 0.072 inch, channels 1 to 17 -- 0.078 inch

PROBLEM 9 Hot spot, hot track, peak power element. Channel 20 -- 0.175 inch, channel 19 -- 0.083 inch, channel 18 -- 0.073 inch, channel 19 has a 0.006 inch dimple at 28.75 inches below the top of the fuel plates.

Complete results of the calculations are listed in Table V-1-5. The maximum stress was 3180 psi in plate 18 at a plate surface temperature of 427°F; the yield strength of 6061-0 aluminum is 5600 psi at 427°F. The maximum stress in plate 19 was 2747 psi at 369°F; the yield strength is 7400 psi at 369°F.

The deflection of plate 19 due to differential pressure acting across the plate was found to be only slightly larger than the induced deflection of plate 18, and since the difference in plate 18 and 19 temperatures is small, it is postulated that the differential deflection between plates 18 and 19 will be negligible for all reactor operating conditions. A similar argument is used for the remaining channels.

2.43 Axial Thermal Stress Analysis. Plates 18 and 19 were analyzed for axial thermal stress at two conditions: (a) the high peak power element, and (b) the high peak power element, hot spot, hot track factors included. Each plate was considered to be integral with its proportional part of the side plate, the resulting unit being free to expand axially, but with the ends restrained against rotation. Plate temperatures were calculated from IBM computer program MACBRE 40.0041 and PDQ 40.0018; side plate temperatures were taken as 140°F. Stresses were calculated from the equation:

$$\sigma(x) = \frac{1}{A} \sum_{i=1}^{14} \left[\Delta T(y)_i \right] \left[E_i \right] \left[\alpha_i \right] - E_x \alpha_x \Delta T(y)_x$$

where

$\sigma(x)$ = axial plate or side plate stress at any point in psi

A = area of fuel plate and side plate -- in.²

i = integration regions as shown in Figure V-1-9

$\Delta T(y)_i$ = average temperature increase in regions i -- °F

E_i = modulus of elasticity of region i -- psi

α_i = coefficient of thermal expansion in region i -- in./in./°F

E_x = modulus of elasticity at point of stress calculation -- psi

α_x = coefficient of thermal expansion at point of stress calculation -- in./in./°F

$\Delta T(y)_x$ = increase in temperature at point of stress calculation -- °F.

The following assumptions were made in the analysis:

- (1) There is no slippage of the swaged attachment of fuel plates to side plates

TABLE V-1-5

THERMAL AND PRESSURE AZIMUTHAL STRESS AND PLATE DEFLECTION

Problem Number	ΔP Channel 19 to 20 (psi)	ΔP Channel 18 to 19 (psi)	ΔT Plate 19 to Rest (°F)	Center-Span Plate 19 Pressure Stress Outside/Inside (psi)	Center-Span Plate 19 Thermal Stress Outside/Inside (psi)	Center-Span Plate 19 Combined Stress Outside/Inside (psi)	Center-Span Plate 18 Total Stress Outside/Inside (psi)	Yield Stress 6061-O Al at Plate Temperature (psi)	Vertical Pressure Deflection Plate 19 to Channel 20+ (in. x 10 ³)	Vertical Thermal Deflection Plate 19 to Channel 20+ (in. x 10 ³)	Combined Vertical Deflection Plate 19 to Channel 20+ (in. x 10 ³)	Vertical Deflection Plate 18 Due to ΔP_{19} to Channel 20+ (in. x 10 ³)	Location Below Top of Plate (in.)
1	-1.24	+0.25	68	330 C 312 T	990 T 1213 C	660 T 901 C	134 C 163 T	7900	-0.55	+2.51	+1.96	-0.51	44.5
2	-1.54	+0.38	89	408 C 371 T	1353 T 1670 C	945 T 1299 C	157 C 193 T	6900	-0.69	+3.72	+3.03	-0.64	44.5
3	-2.18	+0.17	27	580 C 532 T	393 T 482 C	187 C 50 T	252 C 301 T	7900	-0.96	+0.96	0	-0.90	44.5
4	-2.41	+0.27	47	638 C 581 T	715 T 882 C	77 T 301 C	265 C 318 T	7000	-1.07	+1.97	+0.90	-1.00	44.5
5	+4.05	0	57	1075 T 988 C	830 T 1016 C	1905 T 2004 C	496 T 575 C	8000	+1.78	+2.10	+3.88	-1.66	14
6	+4.47	0	89	1184 T 1077 C	1353 T 1670 C	2537 T 2747 C	522 T 617 C	7400	+1.99	+3.72	+5.71	+1.86	14
7	-0.93	+0.24	77	247 C 227 T	1120 T 1374 C	873 T 1147 C	98 C 120 T	8000	-0.41	+2.84	+2.43	-0.87	44.5
8	-1.25	+0.37	96	331 C 301 T	1460 T 1804 C	1129 T 1503 C	123 C 154 T	6500	-0.56	+4.02	+3.46	-1.17	44.5
9	0	+5.15	77	0 0	1170 T 1445 C	1170 T 1445 C	3180 T 2650 C	5600	0	+3.22	+3.22	0	28.75

FIG. V-1-9 FUEL PLATE THERMAL STRESS INTEGRATION REGIONS.

- (2) The fuel plate material is homogenous 6061-0 aluminum
- (3) The elastic limit is not exceeded
- (4) The side plate temperature is 140°F
- (5) The initial temperature of fuel plates and side plates is 70°F.

Maximum stresses for the peak power element were: (a) 16,960 psi tension in the side plate adjacent to plate 19, (b) 14,450 psi compression at the center of the fuel in plate 19, (c) 11,450 psi tension at the surface of plate 19, and (d) 7300 psi compression at the surface of plate 19. Maximum stresses for the peak power element, hot spot, hot track, factors included were: (a) 22,500 psi tension in the side plate adjacent to plate 19 (b) 18,800 psi compression at the center of the fuel in plate 19, (c) 16,800 psi tension at the surface of plate 19, and (d) 10,200 psi compression at the surface of plate 18. Table V-1-6 lists plate temperatures and stresses for each region of plate 19.

2.44 Discussion of Results. The theoretical analysis shows that the azimuthal fuel plate stresses developed by hydraulic pressure differentials and thermal differentials will be much less than the yield strength of 6061-0 aluminum at all reactor operating conditions for elements meeting design specifications. The analysis indicates, however, that axial thermal stresses will exceed the yield strength of the fuel plates at some points across the plates, particularly near the center of the fuel.

Four factors have influenced the conclusion that axial yielding will not affect the operation of the fuel element:

- (1) Axial thermal stresses are not additive with axial stresses and will not contribute to plate deflections
- (2) Regions where the axial yield stress is exceeded are small
- (3) Yielding reduces thermal stresses

TABLE V-1-6

AXIAL THERMAL STRESS DISTRIBUTION PLATE 19
 PEAK POWER ELEMENT AT 25 INCHES BELOW TOP OF PLATES
 (Channel 20 -- 0.113 inch,
 Channel 1 -- 0.108 inch, Channel 2 to 19 -- 0.078 inch)

Region i[a]	Average Clad Surface Temperature $T(y)_i$ (°F)	$E_i \times 10^6$ (psi)	$\alpha_i \times 10^{-6}$ in./in./°F	$\Delta T(y)_i$ (°F)	$E_i \cdot \alpha_i \cdot \Delta T(y)_i$ (psi)	$\sigma(x)_t^{[b]}$ (psi)	$\sigma(x)^{[c]}$ Stress at Outside Clad (psi)
1	140	9.85	13.9	70	8,890	25,850	+16,960 T
2	208	9.6	13.2	138	17,500		+ 8,350 T
3	327	9.1	13.35	257	31,200		- 5,350 C
4	348	8.9	13.4	278	33,100		- 7,300 C
5	338	9.0	13.35	268	32,200		- 6,350 C
6	321	9.1	13.35	251	30,500		- 4,650 C
7	301	9.2	13.3	231	28,300		- 2,450 C
8	281	9.3	13.25	211	26,000		- 150 C
9	267	9.3	13.2	197	24,200		+ 1,650 T
10	256	9.4	13.2	186	23,100		+ 2,750 T
11	249	9.4	13.2	179	22,200		+ 3,650 T
12	234	9.5	13.15	167	20,900		+ 4,950 T
13	183	9.7	13.1	113	14,400		+11,450 T
14	140	9.85	12.9	70	8,890		+16,960 T

[a] See Figure V-1-9 for region locations.

$$[b] \sigma(x)_t = \frac{1}{A} \sum_{i=1}^{14} [\Delta T(y)_i] [E_i] [\alpha_i].$$

$$[c] \sigma(x) = \sigma(x)_t - E_x \alpha_x \Delta T(y)_x.$$

(4) Thermal distortion tests, reported in this section, showed no significant plate rippling.

In addition to the above factors, the ATR fuel elements will be examined during the start-up program at several operating levels to ensure satisfactory performance.

2.5 Irradiation Test Results

The Al-U₃O₈ system has been used successfully in the Argonaut Reactors, in the Puerto Rico Nuclear Center Test Reactor, and is scheduled for use in the High Flux Isotope Reactor at Oak Ridge. Prior irradiation experience has been low temperature (less than 300°F) tests of fuel plate samples and full-sized

MTR-type fuel elements. ORNL has successfully irradiated to 65 weight percent U_3O_8 (20 percent enriched) MTR-type fuel elements in the lattices of the MTR and ORR, to burnups of 30 and 50 percent U-235, respectively. Samples of 63 weight percent U_3O_8 (20 percent enriched) fuel plates were irradiated to 40 percent U-235 burnup. Additional irradiations by Phillips Petroleum Company and ANL, using 22 weight percent U_3O_8 (93 percent enriched) with natural boron and 37 weight percent U_3O_8 (19 percent enriched), respectively, to 27 percent U-235 burnup, were successful.

An irradiation program has been started to test specimens of ATR fuel composition at conditions expected during operation.

An initial test to examine irradiation behavior of 35 and 44 weight percent U_3O_8 (fully enriched) -- X8001 aluminum matrix containing 0.19 weight percent B_4C was performed in the ETR lattice. The cladding was X8001 aluminum 10 percent cold worked. The samples were anodized to provide a thermal barrier yielding fuel temperature expected in the ATR. All samples were dimensionally stable for exposures of 6 to 7×10^{20} fissions/cc [9]. This burnup corresponds to the average ATR core midplane burnup expected during a 17-day cycle. After 1.1×10^{21} fissions, the above samples had blistered although no fission product release was detected in the reactor process water. Assessment of a realistic final surface temperature was complicated by uncertainties in the anodized coating thickness and thermal conductivity.

The irradiation program for U_3O_8 -aluminum system has required the installation of a once-through loop facility in the ETR. This facility will allow controlled irradiations of the U_3O_8 -Al system to be performed at temperatures, flow, and fissioning rate expected in the ATR.

2.6 Corrosion Tests

Corrosion of ATR fuel element cladding can affect system performance in two ways:

- (1) Penetration of the clad by corrosion can lead to release of fission products
- (2) Low conductivity oxide layer buildup can increase the metal temperature, reduce the strength properties, cause fuel swelling and distortion, and lead to failure.

In order to evaluate the ATR fuel element design, extensive corrosion testing of ATR fuel element cladding was performed out-of-pile [10].

Results of these tests are as follows:

- (1) Corrosion rate increases by about a factor of 2.7 as the pH is increased from 5.0 to the range. 5.7 to 7.0.
- (2) Corrosion rate is a function of time and water-oxide interface absolute temperature. Based on the test results at pH-5, the relationship can be expressed by the formula $X = 443 e^{0.778 - \frac{8290}{R}}$

where

X = corrosion film in mils,

θ = time in hours,

R = absolute oxide-water surface temperature -- Rankine scale.

(3) Coolant velocity between 25 to 48 fps was not a significant variable except in influencing the initial surface temperature.

(4) Heat flux was not important in the range 1 to 2×10^6 Btu/hr-ft² when all other test conditions were equal. Lower oxide accumulation rates were observed at 0.5×10^6 Btu/hr-ft².

(5) Coolant temperature was not significant except for its influence on specimen surface temperature.

Film stripping occurred when the oxide layer reached about 2 mils in thickness. Some localized attack (in the form of subsurface voids), extending several mils into the metal, was observed when film stripping occurred.

The layer of oxide was alpha aluminum oxide monohydrate boehmite. Thermal conductivity of this material was measured at 1.3 ± 0.2 Btu/hr-ft-°F.

In-pile corrosion testing of samples of ATR fuel plate composition at expected ATR conditions will provide the information necessary to predict ATR fuel element behavior for the first core test program.

Final testing will be performed in the ATR reactor at several power levels. Periodic examination of the elements will be made during first core operation.

It is concluded that the out-of-pile tests indicate the possibility of developing oxide coatings on ATR fuel plates in the range of 1 to 3 mils. In-pile tests will determine expected buildup rates under simulated ATR conditions.

2.7 Chemical Reactions Between Aluminum and U₃O₈ Dispersions

Several chemical reactions between aluminum and U₃O₈ dispersions, classed as "thermite" reactions, have been studied [11]. These exothermic reactions reduce U₃O₈ to UO₂ and UAl_(n) with a maximum theoretical energy release of 1.84 MW-sec/kg U₃O₈ or 92 MW-sec/ATR core.

An experimental program to study the kinetics of the above reactions was undertaken at the Georgia Institute of Technology Engineering Experiment Station. The primary objective was to obtain reaction rate data using ATR fuel plates, thus providing information necessary to evaluate possible reactor safety problems posed by these reactions.

The kinetics were studied by a variety of techniques including differential thermal analysis (DTA), dynamic high temperature X-ray diffraction, post-heating chemical analysis, and metallography.

A major problem encountered in the studies was to determine reaction rates during the heat-up (equilibration) period.

The program results were as follows:

- (1) Thermite reaction conversion for the ATR fuel plate system is less than three percent below 1400°F.
- (2) After a reaction period corresponding to the time period during which the sample temperature equilibrates, the reaction rate is less than 10 percent U_3O_8 reduced per minute.
- (3) The reaction rates during the equilibration period could not be determined by methods currently available at Georgia Institute of Technology.
- (4) The following reaction mechanism was indicated:

As the reaction begins, U_3O_8 is reduced to UO_2 forming a barrier of Al_2O_3 and UO_2 around the U_3O_8 particles. The formation of the reaction barrier is rapid. The rate of conversion to UO_2 is retarded as the thickness of the reaction product layer increases. The thickness of the barrier layer is determined by the conversion rates to UO_2 and UAl . The conversion rate of U_3O_8 to UO_2 is controlled by mass transfer of aluminum to the reaction zone. A volume reduction of 12 and 21 percent occurs from the reduction of U_3O_8 to UO_2 to UAl , respectively, leading to a porous barrier of fine particles. Aluminum movement should, therefore, occur by bulk flow rather than by diffusion.

- (5) The total conversion during the equilibration period was temperature dependent. Maximum conversion during this period was 65 percent $\text{U}_3\text{O}_8 \longrightarrow \text{UO}_2$ and 15 percent $\text{U}_3\text{O}_8 \longrightarrow \text{UAl}$.

Studies by ANL showed that 39 weight percent U_3O_8 -Al systems exhibited self heating when heated above 1472°F. Self heating rates of 200°C/sec were found when specimens were heated in an oven to 2280°F. This corresponds to a reaction rate of 20 percent per second U_3O_8 reduction to UO_2 .

Presintered cermets showed higher self-heating effects than unsintered material.

Ignition of ATR fuel material does not occur. Sintered specimens containing 75 to 85 weight percent U_3O_8 would ignite.

Experiments showed that reaction rates are a direct function of U_3O_8 particle size.

Several experiments showed that ATR fuel matrix material retains its shape under load above 1400°F. These tests did not simulate ATR environment.

It is concluded that thermite reactions can occur in ATR fuel under certain conditions such as a flow blockage accident, and the energy release involved (although not large or explosively fast) must be considered in evaluating the possibility of metal-water reactions.

2.8 Summary

The ATR fuel element design has been based on extensive research and development work that has included:

- (1) Fabrication techniques
- (2) Hydraulic development including pressure deflection tests and associated analytical program, hydraulic buckling tests, thermal distortion testing, and hydraulic proof tests of the final design.
- (3) Irradiation tests
- (4) Corrosion tests
- (5) Chemical reaction studies.

In addition, thermal stress analysis was made of the final design.

Although the results of all tests are not completed, the following conclusions are made:

- (1) Acceptable commercially fabricated fuel elements have been made.
- (2) The ATR fuel element hydraulic design is satisfactory for full power operation provided the fuel plate-side plate joints are made according to specified technique. Hydraulic proof testing provides assurance that poor fabrication techniques are easily detected. Several minor modifications, such as the comb design, have been specified for the first ATR core purchase order.
- (3) The irradiation behavior of the fuel under expected ATR conditions will be more adequately defined following completion of present irradiation program.
- (4) The corrosion behavior of the fuel clad under expected ATR conditions will supplement out-of-pile results following completion of present irradiation program. Inspection procedures during approach to power tests, the planned stepwise approach to power, and full power operation of initial cores will show up any significant corrosion problems.
- (5) Exothermic chemical reactions between aluminum and U_3O_8 are not significant below 1400°F. Above this temperature chemical reactions can occur. The reaction rate is not explosive and ignition has not been observed at U_3O_8 concentrations to be used in the ATR fuel. The effects of these reactions are considered in Section XVI-3.

3. REFLECTOR AND MODERATOR

3.1 Description

Light water is used as the core moderator and coolant. The eight metallic beryllium segments surrounding the core serve as a reflector. The reflector is 51 inches high and extends 2 inches above and 1 inch below the four-foot active core. Each segment contains two beryllium drums, 7-1/4 inches in diameter, which have hafnium absorber plates covering 120 degrees of the surface as shown in Figure V-1-10.

3.2 Cooling of Reflector

The reflector is cooled by an array of full length 5/16 inch-diameter holes bored through the blocks and by the exterior surface annuli of the blocks. The coolant arrangement was designed to limit the maximum temperature in the reflector to 250°F. Spacer pads are included on each exterior surface of either the reflector or the mating part to ensure that the required minimum coolant gap is maintained. The drums and the cores in the reflector have 0.125-inch radial clearance. Adequate cooling has been provided in the annulus around the drums for the case of minimum tolerances, thermal bowing, and distortion caused by differential irradiation induced growth of the beryllium in both the blocks and drums.

3.3 Stress Analysis and Clearances

Stress analysis indicates that the combined stress due to thermal gradients, and strain caused by irradiation growth for 300-days operation should not exceed 25,100 psi.

The radial clearance between the reflector core and the control cylinder is 0.125 inch. The sum of maximum tolerances, thermal bowing, and distortion caused by irradiation growth (for 300-days irradiation at power) is 0.076 inch as noted in Table V-1-7. Thus, a minimum radial clearance of 0.049 inch exists after 300-days operation to ensure adequate cooling and clearance for operation.

A gross failure of any part of the reflector would not inhibit the scram action of the safety system because safety rods are completely enclosed by the flux trap baffles in and below the reflector region of the core. A failure of the beryllium could make one or more of the reflector control cylinders inoperative, but it is considered incredible that all cylinders would become simultaneously inoperative.

In the event the hafnium absorber plates on the control cylinders are released from the beryllium, they are still retained in the reflector region because the bore is closed at the bottom. Hence, no significant change in reactivity could be caused in the reactor by such a failure.

The reflector is made of vacuum-cast, hot pressed beryllium. It is radio-graphed to inspect internal integrity, and liquid penetrant inspected for surface quality, including the surface of the coolant holes.

3.4 Corrosion and Radiation Effects Studies

Beryllium corrosion rates were studied out-of-pile under environmental conditions expected during ATR operations. Long-term (4000 hours) tests

TABLE V-1-7

VARIATION IN RADIAL
CLEARANCE BETWEEN CONTROL
CYLINDER AND REFLECTOR AT
CORE MIDPLANE

	<u>Summation</u>
Tolerances and Clearances (0.038 at top, 0.029 at bottom)	0.034 inch
Thermal Bowing	
Control Cylinder	0.021 inch
Reflector	0.006 inch
Shaft Slope	0.002 inch
Irradiation Distortion at 300 days	
Control Cylinder	0.006 inch
Reflector	0.006 inch
Shaft Slope	0.001 inch
Total	0.076 inch
Nominal Clearance (radial)	0.125 inch
	-0.076 inch
Minimum Radial Clearance	0.049

showed an overall corrosion rate of 4.4 to 5.9 mpy. Unpredictable short-term rates as high as 8.5 mpy were obtained. Principal parameters effecting corrosion were found to be: pH, temperature, exposed surface area, loop volume, ion exchange flow, velocity, and time. An unknown parameter(s) was believed to have caused the higher short-term corrosion rates [12].

Stressed beryllium specimens were not susceptible to stress corrosion in the ATR environment [12].

Radiation damage to beryllium is generally attributed to displacements produced by fast neutrons and by particles (after slowing down) and transmutations by thermal neutron absorption. Diffusion of the gaseous atoms to dislocation centers can introduce stresses which lead to deformation and failure [13, 14, 15, 16, 17]. Cracking

and deformation of several MTR beryllium reflector pieces has been reported [18]. Measured strain values of 0.0021 inch/inch are reported for integrated fast neutron fluxes (>1 MeV) of 1×10^{22} nvt.

A surveillance and materials irradiation program is expected to ensure the integrity and indicate the useful life of the ATR beryllium reflector. Periodic measurements of the beryllium-fuel annulus spacing using special gauges will be made. Previously irradiated beryllium specimens will be located in ATR positions of high fast flux to provide a dose-damage history of the material in advance of possible reflector damage.

4. CONTROL AND SAFETY ELEMENTS

4.1 General Description

The ATR utilizes a safety system, two shim systems, and a regulating system. The safety system consists of five safety rods located in the north, southeast, south, southwest, and northwest lobes. The outer shim system consists of 16 rotating beryllium cylinders with attached poison inserts. The second shim system consists of six, small diameter neck shim rods located in each of four neck shim rod housing arms. The regulating system is composed of two of the 24 neck shim rods.

4.2 Safety Rods

Each vertically moving safety rod assembly is composed of four major components; an aluminum guide tube, poison inserts, flux trap fillers, and hold-down-latch. The seven-foot-long aluminum guide tube is the structural member

of the assembly guiding and supporting the other components. The guide tube surrounds an experimental loop tube and its insulating can. The four poison inserts of each rod are 90-degree segments of curved hafnium plate 0.25 inch thick and 36 inches long. Each segment is inserted in longitudinal slots in the upper section of the guide tube. The aluminum flux trap fillers are installed in the same guide tube slots in the follower section below the poison inserts. A variety of flux trap filler pieces can be used to provide a range of aluminum-water ratios in the flux trap annulus surrounding an experiment. The hold-down latch consists of four, self-locking arms to retain the poison and filler inserts. The poison and filler inserts can be removed remotely by special handling tools without disturbing the experimental loops. Adequate cooling is provided by flow through the annuli formed by the flow baffle, guide tube, and experiment insulation can. The safety rods are raised and cocked above the core prior to reactor start-up. This positions the poison section above the core (the poison section penetrates three inches below the top of the active lattice to achieve more rapid bite during a scram) and the flux trap filler section in the active area. Fuel is removed in the vicinity of a safety rod prior to removal of the rod.

4.3 Neck Shim Rods

The neck shim rod element consists of a 48-inch fluted, hollow hafnium rod approximately 3/8 inch in diameter and a 48-inch aluminum follower rod 1/4 inch in diameter. The aluminum and hafnium rods are joined by a threaded and pinned joint. This assembly is attached mechanically to a flexible drive cable.

The regulator rod positions are the fourth rods from the center in each of the four neck shim regions. This arrangement provides for the availability of a regulator rod in any of the four neck regions, although only two of the four are equipped with the special servo drive system.

Raised sections on the flutes are spaced along the length of the hafnium rods to maintain an adequate coolant annulus between the hafnium rod and the aluminum housing. Inconel X wearsleeves are installed in the neck shim housing for those positions which have regulator rods.

4.4 Outer Shim Drums

The outer shim poison insert is a 120-degree segment of 1/4-inch-thick curved hafnium plate composed of six axially aligned pieces, each approximately 7-5/8 inches long which are retained against the beryllium cylinders by two aluminum extrusions inserted into slots in the cylinders.

4.5 Hafnium Performance

The poison sections of the neck and outer shim control elements are wrought hafnium. The hafnium operates at a maximum surface temperature of 240°F receiving a maximum possible fast integrated flux (greater than 1 MeV) per reactor cycle of about 6.7×10^{20} n/cm² in the neck shim elements, and 1.8×10^{20} n/cm² in the outer shim elements. Since the shim elements are not positioned permanently in the core, they will experience actual dosages somewhat less than above depending on the control requirements.

Corrosion testing of the hafnium at ATR conditions shows that no corrosion problems exist for the ATR control element.

Irradiation data on arc-melted, crystal bar hafnium shows that the tensile properties are not seriously impaired by an integrated fast flux (greater than 1 MeV) of 2×10^{21} n/cm². The data on the elongation, reduction in area, yield strength, and tensile strength, presented in Table V-1-8, indicate that the properties would be structurally acceptable after higher irradiation.

TABLE V-1-8

HAFNIUM TENSILE PROPERTIES[25]

	Yield Strength (psi)		Tensile Strength (psi)		Elongation (percent)		Reduction in Area (percent)	
	77F	300°F	77F	300°F	77F	300°F	77F	300°F
Before Irradiation	33,600	28,700	65,000	53,200	23	29	37	44
After 2×10^{21} n/cm ²	67,000	57,200	81,900	63,200	19	23	25	37

V-2. NUCLEAR CHARACTERISTICS

The ATR core design has been developed from extensive critical experiments and a comprehensive analytical program.

The ATR critical experiment (ATRCE) a mock-up of the ATR core design, provided initial experimental information to develop analytical techniques, yielded data for safety analyses, and ensured design requirements were satisfied.

The analytical program developed two dimensional, time-dependent nuclear data to ensure core performance during a fuel cycle, provided the data necessary to fix fuel loading, and supplied information for various safety analyses.

The ATR critical facility (ATRC), a mock-up of the ATR final design, is providing verification of ATR core parameters.

1. REACTOR PHYSICS

1.1 Reactivity Distribution and Control Worth

Table V-2-1 shows the reactivity distribution and the amount of movable control available at the beginning and end of a fuel cycle. A comparison between the calculated (Turbo) and measured values (in the ATRC) is also presented. The measured values were based on a 35 kg (U-235) core corrected to 39 kg, whereas the calculated values used a 39 kg core model.

A Wigner-Wilkins spectrum was used in the fuel region and a Maxwellian spectrum was used in all other regions.

The reference core conditions used in the calculations based on design requirements are shown in Table V-2-2.

Reactivity losses during the fuel cycle result from xenon, samarium, stable fission products, experiment burnup, and fuel depletion. The high flux level in the ATR results in rapid xenon buildup approaching saturated equilibrium concentration. Fuel depletion during a normal (17 day) fuel cycle amounts to 13.7 atom percent.

Reactivity gain occurs from the depletion of the burnable poison (boron). The maximum effective multiplication factor is always less than at beginning of core life. About 56.3 percent of the initial B-10 concentration is consumed during a normal fuel cycle.

The net loss in the effective neutron multiplication factor is 0.061 during the 17-day cycle. The final value of k_{eff} at the end of lifetime was calculated (Turbo) to be 1.057.

The ATRC results indicate the worth of the experiments is a very significant source of reactivity.

TABLE V-2-1

REACTIVITY DISTRIBUTION AND MOVABLE CONTROL

Conditions	CALCULATIONS			MEASUREMENTS	
	Calculations		(39 kg Core) 17 Days $\Delta\rho$	ATRC (35 kg core corrected to 39 kg) 0 Days	
	ρ [a]	$\Delta\rho$		ρ	$\Delta\rho$
Cold clean ^[b] , no burnable poison	0.170			0.209	
Worth of burnable poison		-0.074			-0.086
Cold clean, with burnable poison	0.096			0.123	
Temperature effect from ambient to operating conditions		+0.006 ^[c]		+0.006 ^[d]	
Hot operating with burnable poison	0.000			0.000	
Movable control required		-0.102	-0.054		-0.129
Movable control required with power split		-0.102			-0.110
<u>Poisons</u>					
B-10		-0.074	-0.035		
Xe		0	-0.040		
Sm		0	-0.006		
Fission Products		0	-0.016		

Available Movable Control

	$\Delta\rho$	$\Delta\rho$	
		Maximum Worth	Minimum Worth
Outer Cylinder Shims	0.061	0.081	0.058
Neck Shims	0.048	0.071	0.065
Shim Interaction	0.030		
Total	0.139	0.136	
Total Movable Control Used	0.102	0.110	
Total Excess Movable Control	0.037	0.026	

$$[a] \rho = \frac{k_{eff} - 1}{k_{eff}}$$

[b] Clean condition is defined by all rods withdrawn.

[c] This value includes the positive effect of heating nine water loops from 68 to 600°F. Excluding this positive effect, the core exhibits a negative effect greater than -0.003 $\Delta\rho$.

[d] Not measured.

TABLE V-2-2

REFERENCE CORE CONDITIONS

Fuel Loading	39 kg U-235 uniformly loaded
Boron Loading	165 grams natural boron
Neck Shim Rods	22 neck rods fully inserted
Neck Regulating Rods	2 regulator rods -- one in each low power lobe
N-16 Monitor Tubes	2 N-16 monitors -- one in each of the water holes in the central flux trap that lie on a line with the low power lobes
Fixed Shims	4 fixed shims -- one located in each of the water holes in the central flux trap that lie on a line from the center to each outer lobe flux trap center
Neck Surveillance Samples	Stainless steel samples
Reflector Surveillance Samples	Stainless steel samples
Capsules in Outer Reflector	100% aluminum
Inner Lobe High Power (A-1) Flux Trap ^[a]	78% water, 22% Al
Outer Lobe (A-5) Flux Trap ^[a]	20% water, 80% Al
Inner Lobe Low Power (A-5) Flux Trap ^[a]	30% water, 70% Al
Central Lobe (A-3) Flux Trap ^[a]	40% water, 58% Al, 2% Hf
Outer Cylinder Shim Position Adjacent to Low Power Lobe	Fully inserted
Outer Cylinder Shim Position Adjacent to High Power Lobe	Rotated out 100 degrees

[a] A-1, A-3, and A-5 designate experiments that have fuel densities in the ratio of 1, 2, and 3, respectively, and simulate experiment design criteria.

1.2 Movable Control

Two sources of shim control are used in the ATR. The neck regions between the lobes contain small cylindrical poison rods to control (or shim) some of the excess reactivity of the system and to control the center region power level. Rotating outer shim control cylinders in the reflector, acting in conjunction with the neck shim rods, provide the poison to accommodate power shifts and to control the remainder of the excess reactivity. Table V-2-1 shows the reactivity held in each type of shim control.

Symmetrical shimming operations, such as withdrawing neck shim rods (one at a time) in opposite lobes, simplifies the problem of maintaining a preset

power distribution. The worth of a single neck shim rod does not depend on whether it is pulled by itself or as part of a symmetrical pattern. A symmetrical neck shim program can be used if desired.

The rotating outer shim control cylinders have a calculated total reactivity worth of $0.061 \Delta\rho$. Since the outer shim cylinders are connected mechanically in pairs, two always rotate together. Satisfactory operation is achieved without the added complexity of individual outer shim cylinder operation. Each lobe has two pairs of outer shim cylinders or a total of four cylinders in each lobe.

Combined neck shim rod and outer shim cylinder worth is greater than the sum of the individual worths as shown in Table V-2-1. When the outer shim cylinders are inserted, the importance of the neck region is greater than when the cylinders are withdrawn. Conversely, when the neck shim rods are in, the outer shim cylinders assumed a higher importance which enhances their worth. This interaction adds $0.030 \Delta\rho$ to the calculated shutdown margin when all controls are in. Some degree of interaction is always present regardless of the control positions.

The worth of the regulator rod ranges from 0.002 to $0.004 \Delta\rho$ depending upon time of cycle. The power deficit for the ATR, assuming the loops are at or near operating temperature, is $0.0038 \Delta\rho$. Accordingly, the reactor will require only a moderate amount of shim action to increase the power from 1 to 100 percent.

1.3 Safety Rods

The ATR safety rod worths have been measured in the ATRC. The method used was based on period measurements using the outer shim rods and balancing with safety rods. Period values were converted to reactivity by the inhour equation. The results are shown in Table V-2-3 for the two extreme core power arrangements expected in the ATR. Based on these measurements and a value of β_{eff} of 0.75 percent, a five rod scram in the ATR would insert 6.8 to 7.5 percent negative reactivity. Assuming the most reactive rod stuck, a four rod scram would insert 4.6 to 5.1 percent negative reactivity.

The poison sections used in the measurements were cadmium. Since the ATR safety rod poison sections are hafnium, a test will be made in the ATRC to determine hafnium-cadmium equivalence. It is expected that the hafnium rods will be worth more than the cadmium rods because of resonance factors.

The ATR safety rod integral worth curve is shown in Figure V-2-1.

1.4 Temperature and Void Coefficients

Temperature and void coefficients for the ATR core are based on two dimensional calculations. The results are shown in Tables V-2-4 and -2-5.

The reactivity addition due to total voiding the beryllium reflector at operating power is $+0.0071$.

The end of life fuel region temperature coefficient (operating at N_F) is $-0.97 \times 10^{-4} \Delta\rho/^\circ\text{F}$.

TABLE V-2-3

ATRC SAFETY ROD WORTHS -- DOLLARS
(Cd -- poison pieces)

Safety Rod Number	40-50-60 Power Split ^[a]		Balanced Core	
			All Outer Shims at $\approx 80^\circ$ All Neck Shims In	
1	1.9	Rods 1 and 5 ganged	3.9	1.3
2	2.4			2.2
3	1.4	Rods 3 and 4 ganged	3.6	2.1
4	2.4 ^[b]	Rod 2 (= 4)	2.4	2.2
5	1.9			1.3
Total Shutdown ^[c]	10.0		9.9	9.1

[a] Outer shims N12, W34, S12, and E34 $\approx 100^\circ$. Outer shims N34, W12, S34, and E12 inserted. Neck shims 5 and 6 in each neck withdrawn, all others inserted initially.

[b] For the purpose of calculating each rod worth, all individual worth curves were assumed but not measured to have the same shape as that of this rod. During the calibration of this rod, the reactivity balance was maintained by withdrawing the ganged sets of #4 and #3 neck shims in this order, terminating with the #4 set completely withdrawn and the #3 set at ≈ 25 inches for complete insertion of #4 safety rod.

[c] Total shutdown obtained by summing individual rod worth.

FIG. V-2-1 INTEGRAL SAFETY ROD WORTH

Total reactor cold to hot operating condition is + 0.006 -- including the positive effect of heating loops to 600°F.

A series of tests was performed in the ATRC to investigate voiding in the A-1 flux trap because of the high positive value of the coefficient that was calculated (+ 0.0149).

The objective of the tests was to determine the reactivity effects of voiding an inner A-1 flux trap and experiment pressure tube as a function of shim position.

TABLE V-2-4

BEGINNING-OF-LIFE TOTAL TEMPERATURE COEFFICIENTS

Perturbed Region	Coefficient Hot (Operating at N_F)	$\Delta\rho_t/^\circ\text{F}$ Cold (68°F)
Fuel	-1.37×10^{-4}	-1.06×10^{-4}
Fuel Assuming a Maxwellian Spectrum	-1.49×10^{-4}	
Fuel Plus Side Plates and Surrounding Water Channels	-1.12×10^{-4}	
Neck	$< +1.00 \times 10^{-5}$	$< +1.00 \times 10^{-5}$
Beryllium Reflector	$+0.14 \times 10^{-4}$	$< +1.00 \times 10^{-5}$
Nine Experiments	$+0.15 \times 10^{-4}$	$< +1.00 \times 10^{-5}$
A-1 Flux Trap (all water)	$+0.12 \times 10^{-4}$	$+0.1 \times 10^{-4}$
A-5 Flux Trap	$< +1.00 \times 10^{-5}$	$< +1.00 \times 10^{-5}$
A-3 Flux Trap	$+0.1 \times 10^{-4}$	$< +1.00 \times 10^{-5}$
All Nine Flux Traps	$+0.38 \times 10^{-4}$	$+0.28 \times 10^{-4}$

TABLE V-2-5

BEGINNING-OF-LIFE VOID COEFFICIENTS

	Coefficient Hot (operating at N_F)	$\Delta\rho/\Delta V/V$ Cold (68°F)
Fuel	-0.209	-0.203
A-1 Experiment (excluding flux trap)	+0.002	
A-5 Experiment (excluding flux trap)	+0.001	
A-3 Experiment (excluding flux trap)	+0.002	
A-1 Flux Trap (high power lobe)	+0.0149	
A-3 Flux Trap (high power lobe)	+0.0085	
A-5 Flux Trap (high power lobe)	+0.0033	

The procedures used in the tests were as follows:

- (1) To determine the effect of shim positions on the reactivity values, measurements were done with three shim configurations: (a) outer shims inserted and 18 neck shims withdrawn; (b) all outer shims at 80° and all neck shims inserted; and (c) outer shims surrounding the lobe with the void device and the diametrically opposite lobe at 90°, other outer shims inserted and only two neck shims withdrawn. This latter shim configuration is that for a 40-50-60 MW power split with the 60 MW lobe surrounding the void device. The measured value of reactivity for a given void was found to be dependent upon shim configuration and was largest with the shims in the power split configuration.
- (2) The experiment flow tube was initially voided and the reactivity change measured for the various shim configurations.
- (3) From 0 to 40 percent flux trap voiding, the water in the filler region was displaced with air. A device consisting of a ring of vertical aluminum tubes was placed in the filler region. The metal-to-water ratio in the filler region with this device in place was the same as if the A-1 type filler were in place. Various void fractions were introduced by symmetrically distributing air filled tubes among water filled tubes within the ring.
- (4) From 40 to 60 percent voiding, the water in the filler region was displaced with aluminum rods in the interstitial spaces of the fully voided ring of tubes. The approximation was made that, except for some absorption, the aluminum acts as air would and that such absorption is directly proportional to the volume of

aluminum present. To determine the correction for this absorption, the 20-percent-void case was done with twenty percent of the water displaced with air and repeated with twenty percent of the water displaced with aluminum. The observed reactivity difference was interpreted as being due to the absorption of the aluminum and was extrapolated on the basis of aluminum volume to provide a correction for all measurements requiring aluminum additions.

Because the safety rods occupy the positions containing A-1 type experiments, the NE flux trap was used for these measurements. It was converted to an A-1 type flux trap except that the fuel loading was left at 80.3 g U-235 rather than the 20.5 g U-235 normally present. This difference is expected to make the measurements slightly conservative, although the difference would be expected to be small.

Although the results indicate that the reactivity is a linear function of the percent void up to 60 percent, it would be expected that beyond this last point the curve will tend to flatten out. Additional work will be performed to determine the curve with greater than 60 percent void.

Figure V-2-2 shows the results of the measurements. The shim position corresponding to the power split yields a linear extrapolation for 100 percent voiding of an A-1 flux trap of 1.5 dollars. The reactivity, based on a value of β_{eff} corresponding to the outer shim positions, is 1.05 percent. All other allowable shim arrangements yield lower reactivity values for 100 percent voiding.

1.5 Neutron Flux Distribution

Turbo calculations for the ATR at a power level of 250 MW and unbalance lobe power (40-50-60 MW power split) results in the following average fluxes in the fuel elements comprising the three types of lobes.

FIG. V-2-2 REACTIVITY EFFECT OF VOIDING AN A-1 FLUX TRAP AND EXPERIMENT FLOW TUBE.

Time-Days	0		17	
	ϕ thermal n/cm ² sec	ϕ epithermal n/cm ² sec	ϕ thermal n/cm ² sec	ϕ epithermal n/cm ² sec
High Power Lobes -- 60MW	2.6×10^{14}	13.5×10^{14}	3.2×10^{14}	13.9×10^{14}
Low Power Lobes -- 40MW	1.6×10^{14}	10.2×10^{14}	2.0×10^{14}	10.4×10^{14}
Central Lobe -- 50MW	2.1×10^{14}	14.6×10^{14}	2.5×10^{14}	14.8×10^{14}

The power generated in each fuel element (quarter core) at time zero is shown in Figure V-2-3. The location and magnitude of maximum peak-to-average midplane core power ratio is also shown.

The experiment fluxes and performance characteristics are shown in Table V-2-6.

The calculated midplane (axial) power profiles for the ATR fuel element containing the core peak is shown in Figure V-2-4. The fuel element sketch in the

* Maximum Peak-to-Average Power Fraction = 2.5

FIG. V-2-3 POWER GENERATED PER ELEMENT AT TIME ZERO.

TABLE V-2-6

ATR EXPERIMENT PERFORMANCE AS A FUNCTION OF LIFETIME

Time (days)	Watts/g U-235 x 10 ⁴ [a]			
	Experiment Number			
	A-1	A-3	A-5 Exterior	A-5 Outer
0	2.213	1.281	1.438	1.069
1/3	2.171	1.287	1.455	1.092
2/3	2.2187	1.293	1.462	1.097
1	2.204	1.299	1.450	1.062
2	2.194	1.316	1.462	1.053
5	2.219	1.309	1.478	1.089
10	2.195	1.268	1.494	1.131
17	2.233	1.321	1.541	1.158
Design Specification	2.100	1.500	---	1.000

Time (days)	kW/ft ² [a]			
	Experiment Number			
	A-1	A-3	A-5 Exterior	A-5 Outer
0	148.2	171.6	288.7	214.6
1/3	144.4	171.7	290.6	218.5
2/3	144.3	171.7	290.5	218.5
1	144.4	171.7	286.7	210.8
2	140.4	171.6	284.7	206.7
5	132.6	163.8	275.0	206.8
10	117.1	148.2	257.4	202.8
17	101.4	140.4	238.0	191.2
Design Specification	140.0	200.0	---	200.0

Time (days)	Midplane Average Thermal Flux x 10 ¹⁴			
	Experiment Number			
	A-1	A-3	A-5 Exterior	A-5 Outer
0	7.63	4.23	4.81	3.59
1/3	7.49	4.33	4.83	3.62
2/3	7.55	4.34	4.79	3.62
1	7.65	4.28	4.77	3.49
2	7.62	4.34	4.78	3.51
5	7.59	4.31	4.84	3.61
10	7.63	4.29	4.93	3.77

TABLE V-2-6 (Cont.)

ATR EXPERIMENT PERFORMANCE AS A FUNCTION OF LIFETIME

	Midplane Average Thermal Flux x 10 ¹⁴			
17	7.85	4.44	5.16	3.86
Design Specification	10.00	4.84	---	3.22
Time (days)	Midplane Average Epithermal Flux x 10 ¹⁴			
	Experiment Number			
	A-1	A-3	A-5 Exterior	A-5 Outer
0	16.22	15.90	16.43	13.53
1/3	15.86	16.31	16.66	13.59
2/3	15.99	16.38	16.92	13.63
1	16.19	16.13	16.75	13.11
2	16.08	16.27	16.83	13.13
5	15.83	16.02	16.65	13.36
10	15.62	15.71	16.44	13.68
17	15.67	15.87	16.10	13.57
Design Specification	10.00	15.00	---	10.00
	Ratio of Epithermal to Thermal Flux			
0	2.13	3.76	3.42	3.77
1/3	2.12	3.76	3.45	3.75
2/3	2.12	3.77	3.53	3.76
1	2.12	3.77	3.51	3.76
2	2.11	3.75	3.52	3.74
5	2.08	3.72	3.44	3.70
10	2.05	3.66	3.34	3.62
17	2.00	3.58	3.12	3.52
Design Specification	1.00	3.10	---	3.10

[a] Midplane values normalized to 175 MeV per fission in the experiments as opposed to 201.6 MeV per fission in the core.

above figure defines the direction used in the profiles. The power depression toward the right edge of the element is caused by the proximity of inserted neck shims. Figure V-2-5 shows the core peak as a function of cycle time for both Maxwellian and Wigner-Wilkins spectrums. The calculations (Turbo) are based on the reference 39 kg core listed in Table V-2-2.

FIG. V-2-4 POWER PROFILES FOR PLATES IN FUEL ELEMENT WITH HOT SPOT.

FIG. V-2-5 CORE PEAK POWER DURING FUEL CYCLE.

Figure V-2-6 shows the midplane power profile measured in the ATRC at the hot spot. The calculated curve above is also shown for comparison. The ATRC experimental arrangement attempted to simulate the model used in the Turbo calculations. Several major differences were the results of the use of a 35 kg core fuel loading in the ATRC compared to the 39 kg core used in the calculations. The number of neck shims required to obtain the same power split used in the calculations was different. The above factors together with homogenization with attendant neglect of self-shielding in the calculations explain the differences in the curves shown in Figure V-2-6.

1.6 Neutron Lifetime and Delayed Neutron Fraction

The measurement of $\ell^*/\bar{\beta}$ (mean neutron lifetime $\div$ effective delayed neutron fraction) was made in the ATRC using two independent techniques: (a) pulsed neutron, and (b) reactor noise analysis. The ratio was evaluated for

FIG. V-2-6 PLATE 19 MIDPLANE POWER PROFILE IN ELEMENT CONTAINING THE CORE HOT SPOT.

eleven different core conditions using the pulsed neutron techniques; the results of these measurements are shown in Table V-2-7. As shown, the value of $\ell^*/\bar{\beta}$ is affected most by the outer shims and, in general, increases as they are withdrawn. An independent determination of $\ell^*/\bar{\beta}$ using reactor spectral noise data also was made for the condition of 78 degrees rotation of the outer shims. The value obtained was 7.02 msec which is in close agreement with the value from the pulsed neutron method.

TABLE V-2-7

VALUES OF $\ell^*/\bar{\beta}$ FROM PULSED NEUTRON EXPERIMENTS IN THE ATRC

Outer Shim Positions (degrees)	Number of Shim Rods Per Neck	Boron Cross Section (cm^{-1})	$\ell^*/\bar{\beta}$ (msec ± 0.10)
0	6	--	6.40
29.2	2	0.0183	6.79
44.0	0	0.0281	7.48
46.0	1	0.0242	8.31
46.15	2	--	7.34
68.05	1	0.0281	7.88
71.0	7	--	8.13
81.58	6	--	8.23
86.58	0	0.0347	7.65
128.8	0	0.0395	9.69
150.3	0	--	9.17

Whether the variation of $\ell^*/\bar{\beta}$ is due to variations in ℓ^* or $\bar{\beta}$ or both is not known. However, qualitatively the following can be said: With the poison in the shims far from the fuel elements (150 degree position), $\bar{\beta}$ will be greater than β (absolute delayed neutron fraction = 0.0064) and would probably be comparable to the value used for the MTR and ETR. For reactors of this type it is customary to assume that Hughes' value of 0.0075 is valid. This value yields a calculated lifetime of 69 microseconds for the shims withdrawn. Now, as the shims insert, or the poison approaches the fuel elements, either ℓ^* decreases or $\bar{\beta}$ increases or a combination of both occurs.

The heavier fuel loading in the ATR will decrease the neutron lifetime. Therefore, measurements will be made early in the ATRC program to determine the new $\ell^*/\bar{\beta}$ values, and attempts will be made to determine relative changes in $\bar{\beta}$ and ℓ^* .

2. KINETIC NUCLEAR CHARACTERISTICS

2.1 General

The reactor uses three types of poison control elements and a burnable poison (B-10) in the fuel elements for reactivity control. The total reactivity worth of all control elements is approximately 20 percent $\Delta\rho$. A maximum reactivity addition rate of 0.1 percent $\Delta\rho$ per second is possible if all control elements could be withdrawn simultaneously. However, this is not allowed under any operating conditions because of control interlocks. Each group of control elements requires eight minutes for a full withdrawal.

2.2 Safety Rods

Five safety rods have a calculated total reactivity worth of about 6 percent $\Delta\rho$. Four of these rods have a withdrawal speed of 4.5 inches per minute. This corresponds to a 0.005 percent $\Delta\rho$ per second maximum rate for each rod. The fifth safety rod is used for fast recovery scram as well as for a relay or electronic scram. Its speed is nine inches per minute and corresponds to a 0.012 percent $\Delta\rho$ per second maximum reactivity rate. When all safety rods are withdrawn in a single group of five, the maximum reactivity rate of this group is 0.032 percent $\Delta\rho$ per second.

2.3 Neck Shim Control Rods

The 24 neck shim rods provide a calculated total reactivity worth of 4.8 percent $\Delta\rho$. These rods are used primarily for fuel burnup shimming and, to a lesser extent, for flux shifting. Their fixed speed is 5.7 inches per minute. The maximum reactivity rate of a single neck shim rod is 0.001 percent $\Delta\rho$ per second. For all neck shims simultaneously in motion, the maximum reactivity rate is 0.02 percent $\Delta\rho$ per second.

Four of these rods are equipped with rack and pinion drive mechanisms, two of which are available for connection to the automatic servo system. The other two operate as shim rods. Of the two servo rods, only one is operated at a time, and this rod can operate over a wide speed range up to fifty (50) inches per second. Based on this maximum speed, the maximum rate from a single regulating rod is approximately 0.6 percent $\Delta\rho$ per second. (Total worth is 0.4 percent $\Delta\rho$.)

2.4 Outer Shim Control Cylinders

The outer shim control cylinders, operating in pairs, serve as shim elements. Approximately 6.1 percent $\Delta\rho$ reactivity worth is provided by these eight pairs of cylinders for use primarily in flux shifting and to a lesser degree in shimming to compensate for fuel depletion, lifetime fission product buildup, and xenon poisoning. Shimming speed is 19.8 degrees per minute. For one pair of cylinders, this speed corresponds to a maximum reactivity rate of 0.0033 percent $\Delta\rho$ per second. The eight pairs operating as a group have a maximum reactivity capability of 0.026 $\Delta\rho$ per second.

The maximum reactivity rates for all control elements are summarized in Table V-2-8.

TABLE V-2-8

CONTROL ROD AND MECHANISM DATA

Type of Rod	Rod Speed	Number of Elements	Rod Worth (percent $\Delta\rho$)	Maximum Rate of Insertion (percent $\Delta\rho$ per second)
Normal Safety	4.5 in./min	4	4.6	0.02
Fast Recovery Safety	9.0 in./min	1	1.4	0.012
Neck Shim	6.0 in./min	22	4.8	0.02
Outer Shim Cylinder	19.8 deg/min	8 pairs	6.1	0.026
Regulating Rod	5 in./sec	(Manual)	0.2-0.4	$\approx 0.060^{[a]}$
	0 to 50 in./sec	(Servo)	0.2-0.4	$\approx 0.600^{[a]}$
Rod Interaction			3.0	0.012
Total all control elements (except regulator rods)				0.090 ^[b]

[a] The regulator rod rates are not included in the total for all rods because of their mode of operation.

[b] The total listed is for ganged operation of all rods. This condition is prohibited by control interlocks.

2.5 Power Regulation

Automatic power regulation requires a faster reactivity rate than is provided in the shim control in order to achieve the power regulation requirements of $\pm 1/2$ percent of full power. The motor pack on the two neck rod mechanisms selected for regulator rod duty permits a variable reactivity rate from zero to approximately 0.6 percent $\Delta\rho$ per second. The regulator rod speed, within this range, is proportional to power error during normal operation. Under no condition will the total regulating rod stroke reactivity worth exceed the delayed neutron fraction. Maximum worth is 0.4 percent $\Delta\rho$.

2.6 Safety Action Response

The time required for scram shutdown by the safety rod elements is determined by delay time associated with the input devices, the safety element release mechanism, and the rod travel. The fastest scram inputs are electronic signals from the nuclear instrumentation. All other scram inputs from a recorder, a sensor, or manual input cause the scram by relay action. The delay associated with the nuclear instrumentation channel for an electronic scram is less than 1 msec, and the scram rod release time is 25 msec or less. A relay scram requires 20 msec for the relay actions, thus a total of 60 msec is required to initiate rod motion after a relay scram signal is received, including a 15 msec amplifier decay time.

Ramp and step reactivity additions have been investigated to determine the relative importance of various nuclear parameters on reactor transient behavior. It was concluded that ramp reactivity additions from rod motion at any power level do not result in transients as severe as those associated with very fast reactivity addition accidents (assumed to be a step).

The ATR response to a 0.6 percent $\Delta\rho$ step addition is shown in Figure V-2-7. The integral of the power is also plotted. The feedback from the various regions of the core is shown in Figure V-2-8. As can be seen, the moderator coefficient produces the greatest feedback effect. The regions of the core which have positive coefficients make a small positive contribution which is nearly masked by the strong negative effect of the moderator coefficient and the safety rods.

The scram level or minor variations of the scram delay time (less than 5 milliseconds) would have very little effect on these step accidents. This can be seen from examination of Figures V-2-7 and -2-8 which show that a 20 percent change in scram level only delays rod motion 2 msec. The displacement of the feedback from the safety rod in Figure V-2-8 by 1 to 5 msec would change the peak power by only a few percent because the net Δk is changed very little.

FIG. V-2-7 ATR RESPONSE TO A 0.6 PERCENT δk -STEP ADDITION AT BEGINNING OF CORE LIFE.

FIG. V-2-8 REACTIVITY FEEDBACK TO A 0.6 PERCENT δk -STEP ADDITION.

Figure V-2-9 shows reactor power decay curves for relay scrams with five rods and three rods. These curves are conservatively based on total worth of 6 percent $\Delta k/k$.

2.7 Xenon Stability

The three possible types of xenon instability examined were interlobe, intralobe, and axial.

Interlobe oscillations are not a potential source of difficulty, because the reflector controls are designed to handle precisely this type of reactivity shift. Xenon effects will undoubtedly be present with the reference power shift between lobes, but they will be handled routinely by the operator along with other nonuniform reactivity lobe changes due to B-10 burnup, fuel depletion, fission product buildup, and experiment changes.

Intralobe oscillation stability has been studied by using a method proposed by R. S. Wick [19]. A single lobe is stable to a radial oscillation within or around the lobe. The combination of lobe size and internal coupling results in conditions which are within the limits of stability.

Of the three types, axial oscillations received the most detailed analysis. An analysis program was conducted to study the axial stability by using three approaches: a space-time dependent calculation solved on a digital computer, a space-time dependent analog computer representation, and the method of Randal and St. John [20], which is based on reactor parameters and flux. Table V-2-9 summarizes the instability threshold lengths determined by the several methods.

The actual bare equivalent height of the reactor is 53.5 inches, 136 cm.

The results of the space-time dependent digital calculations showed that the 48-inch core length is stable. Additional calculations were performed with a core length of 98 inches to provide a comparison of stability of the reference core to a longer one. The results of the longer core show definite instability. Figures V-2-10 and -2-11 show the results of these digital calculations.

The space dependent flux equations were solved on an analog computer. Parametric studies were performed to establish stability as a function of core length. These calculations predicted an xenon instability threshold length of 61.8 inches for the reactor core. An equivalent core length of 53.5 inches was stable. (See Figures V-2-10 and -2-11.)

All methods showed that the reference core length of 48 inches is stable. Xenon stability experiments will also be performed in the ATR during the start-up period.

2.8 Interlobe Coupling

It has been demonstrated in the ATRCE that a high degree of coupling between lobes exists. That is, a reactivity perturbation at one location in the reactor is effective in transmitting the effects to the entire core. The net result is that it is difficult to perturb the flux at any one location without affecting the entire core.

FIG. V-2-9 ATR SCRAM CURVES.

TABLE V-2-9

BARE REACTOR THRESHOLD LENGTHS

Method	Threshold Length	
Randall and St. John	234 cm	92.13 inches
Wick	238 cm	93.7 inches
Two group analytic	155 cm	61.00 inches
Two group analog	157 cm	61.8 inches

This degree of coupling is qualitatively defined by both experimental data and nuclear calculations. In the ATRCE, an analysis of the coupling was made by measurement of the reactor transfer function at several points in the reactor core [21]. To accomplish this measurement, a rotary, self-shielding oscillator was located in experiment position 3 (an inner flux trap position) and detectors were placed in experiment positions 3 and 7. Experiment position 7 is an inner flux trap

position on the opposite side of the core from position 3. A range of reactivity frequencies between 0.02 and 40 cps was investigated. The data from each instrument were analyzed to yield the ratio of $\Delta n/n$ and the phase shift of the response.

The amplitude response of positions 3 and 7 are shown in Figures V-2-12 and -2-13, respectively. The phase response for the two chambers is shown in Figures V-2-14 and -2-15.

The results of this analysis lead to the conclusion that the difference between the amplitude functions of chambers 3 and 7 are the result of statistical deviation from experimental error. The small difference in phase angle response of the two chambers may be caused by drift in the equipment. It therefore appears that experimental data establishes coupling of the lobes of the core.

The large amount of reactivity (approximately 6 percent $\Delta\rho$) required by the movable control system to maintain the 40-50-60 MW power split shows that there is a very large coupling effect between lobes. This is also emphasized by the analysis of the minimum worth of three safety rods. The position of these rods is the northwest, north, and northeast position, and they are capable of producing a reactor shutdown even though they are only on one side of the reactor core.

This evidence shows that all the lobes are coupled together, and a considerable amount of reactivity is required to force the required power shift. Accordingly, no coupling problems are anticipated for the ATR geometry and safety system action will always shut the reactor down.

FIG. V-2-10 INSTABILITY OF EXTENDED LENGTH ATR CORE TO AXIAL XENON OSCILLATIONS.

FIG. V-2-11 STABILITY OF REFERENCE CORE LENGTH TO AXIAL XENON OSCILLATIONS.

FIG. V-2-12 AMPLITUDE RESPONSE -- CHAMBER POSITION 7 -- FUNDAMENTAL.

FIG. V-2-13 AMPLITUDE RESPONSE -- CHAMBER POSITION 3 -- FUNDAMENTAL.

FIG. V-2-14 PHASE RESPONSE -- CHAMBER POSITION 3 -- FUNDAMENTAL.

FIG. V-2-15 PHASE RESPONSE -- CHAMBER POSITION 7 -- FUNDAMENTAL,

V-3. THERMAL AND HYDRAULIC CHARACTERISTICS

1. GENERAL

The thermal-hydraulic characteristics originally specified for the ATR fuel elements were based on the successful performance of similar type fuel elements in the MTR and ETR and, of necessity, included a reasonable extension of this technology. Several areas of uncertainty with respect to the predicted performance of the fuel elements were identified as the design progressed and, accordingly, research and development programs were undertaken to provide assurance that the fuel element design would be satisfactory. These programs are described in this report and included: (a) calculation and measurement of local power densities; (b) fuel element fabrication and inspection; (c) hydraulic performance; and (d) thermal burnout and flow instabilities. Certain phases of these programs are continuing to permit more precise definition of the allowable operating conditions.

Additional thermal burnout work is underway to identify the mode of the phenomenon, ie, local boiling burnout or "boiling disease", particularly with a test section which has a transverse variation in heat flux such as will be experienced in the hot channel of the ATR.

The core parameters resulting from the studies described in this section are summarized in Table V-3-1.

2. CORE HYDRAULIC CHARACTERISTICS

Three primary coolant pumps, connected in parallel, circulate 47,000 gpm under nominal conditions. Flow requirements could be larger due to tolerance stack-up on internals, changes in flux trap geometry to obtain desired experiment performance, and more than 50 percent utilization of auxiliary experimental facilities. Maximum flow expected at an unrecoverable core pressure drop of 100 psi is 52,800 gpm. The three pumps have a combined capacity of 54,000 gpm, and flow is regulated to the required value by a butterfly-type valve. A standby pump is also available.

The pressurized water enters the reactor vessel through two 24-inch nozzles located near the bottom of the reactor vessel. Approximately 95 percent of the coolant flows up along the thermal shields at a velocity of about six feet per second. The remainder is diverted into the large space between the core reflector tank and the inlet flow baffle. This flow maintains the coolant around the spent fuel storage racks at 125°F and facilitates natural circulation cooling of irradiated fuel elements stored in these racks. The coolant from the two flow paths combines in a large plenum above the core internals, reverses direction, and flows down through the fuel elements, flux traps, and other internals. Approximately 27,400 gpm flows through the fuel elements and the remainder is utilized to cool the reactor internals and auxiliary experimental facilities. Table V-3-2 shows the calculated flow distribution through the core region.

TABLE V-3-1

ATR THERMAL HYDRAULIC PARAMETERS

Reactor Power (MW)	250
Inlet Coolant Temperature (°F)	125
Outlet Coolant Temperature (average) (°F)	167
Outlet Coolant Temperature (maximum) (°F)	356
Core Inlet Coolant Pressure (psig)	347
Core Outlet Coolant Pressure (psig)	247
Nominal Channel Coolant Velocity (fps)	47.8
Nominal Core Coolant Flow Rate (gpm)	27,400
Nominal Reactor Coolant Flow Rate (gpm)	47,000
Heat Transfer Area per Fuel Element (ft ²)	33.8
Average Heat Flux (Btu/hr ft ²)	0.587 x 10 ⁶
Maximum Nominal Heat Flux (Btu/hr ft ²) -- heat flow equally divided to each channel	2.23 x 10 ⁶
Calculated Fuel Plate Temperatures at 250 MW and Maximum Lobe Power Split (°F) (at maximum power peaking during fuel cycle for Plate 19)	
	<div>Core Average</div> <div>Maximum Nominal</div> <div>Maximum (hot spot -- hot channel)</div>
Surface	216 350 472
Fuel-Clad Interface	229 415 552
Fuel Maximum	231 423 568
Wall Temperature for Incipient Nuclear Boiling (°F)	462
Wall Temperature Standard Deviation (°F)	55
Wall Temperature at Two Standard Deviations (°F)	460

TABLE V-3-2

FLOW SUMMARY

Flow Region	Nominal Flow Rate -- gpm ($\Delta P_{\text{core}} = 100 \text{ psi}$)
Fuel Elements (Channel #20 -- 0.113 inch and Channel #1 -- 1.08 inch)	27,400
Reflector Block	1,970
Reflector Control Drum	6,310
Neck Region	580
Loop Facilities	5,600
Capsule Holders in Aluminum Reflector	1,040
Capsule Holders in Beryllium Reflector	1,740
Between Reflector and Reflector Tank	1,200
Seal Leakage	520
Siphon Breaker	640
Total	47,000

3. THERMAL-HYDRAULIC ANALYSIS

The basic premise used to establish an allowable operating power level was that the maximum nominal calculated fuel plate surface temperature (not including hot channel or hot spot factors) should be no larger than two standard deviations from the surface temperature at which nucleate boiling will commence. This statistical approach is based on the error propagation method described in the literature [22]. Conventional hot spot-hot channel analysis is also presented for reference.

Measurements in the ATRC and reactor physics calculations (Figure V-2-6) have shown a steep transverse fission rate profile in the vicinity of the core hot spot along the hot channel (channel 19) resulting in the need to determine the degree of transverse mixing of the bulk fluid. It has been calculated for ATR fuel geometry that essentially no mixing occurs when the heat transfer regime is forced convection. Since the effect of nucleate boiling on mixing is not well defined, no transverse mixing is assumed in calculating the surface temperature or the conditions at which thermal burnout may occur.

3.1 Hot Spot-Hot Channel Analysis

The pressure, flow rate, bulk coolant, and wall temperatures over the length of the element were computed using an IBM 7040 program developed for this problem. This program used the entrance and exit coefficients and friction

factors measured in isothermal hydraulic studies. The program allowed pressure equalization at the vents (and the related cross flow) and uneven heat flow from a fuel plate to the two adjacent channels. The heat transfer coefficients were computed from the Dittus-Boelter correlation that has been experimentally verified at ETR conditions and believed to apply to ATR conditions. The resistance to heat flow represented by conduction through the fuel and cladding regions was also incorporated into the program. The calculated azimuthal flux profile as shown in Figure V-2-6 and the radial and vertical profiles measured in the ATRC were used. These profiles were combined to yield the peak flux measured in the ATRC for nominal temperature calculations. The factors shown in Table V-3-3 were used in hot spot-hot track analysis with the exception of the effect of channel tolerances on velocities and the resulting temperatures which were computed directly. All calculations computed a hot track bulk coolant rise and the related higher surface temperature, but computed velocities were based on the average temperature for each channel. Although this produces temperatures above those expected, there is insufficient data currently available to justify calculating the temperatures on a more realistic basis. Figures V-3-1 through V-3-4 show the computed maximum wall (surface) and bulk coolant temperatures for various channel configurations. The hottest wall temperature is always on plate 19 except in the cases shown in Figures V-3-3 and V-3-4 where it switches to plate 18 about two-thirds of the length of the plate.

The maximizing factors used in calculating the maximum wall temperature are summarized in Table V-3-3. The values cited are based on fuel element specifications, measurements from hydraulic and fuel element development programs, reactor physics and thermal calculations, and experience with other test reactor systems.

Several of the factors cited merit further description. Two dimensional heat conduction calculations were made to determine: (a) the decrease in the peak heat flux caused by conduction to the unfueled edges of the fuel plates (fin effect), (b) transverse velocity factor caused by drag near the side plates, and (c) the increase in the peak heat flux resulting from local fuel inhomogeneity. The fin effect calculation resulted in a factor of 0.98 which does not include conduction to the side plates. The transverse velocity factor is 1.00 since the maximum wall temperature occurs about 0.30 inch from the edge of the fuel. This distance is more than adequate to attain average velocity.

Measurements of local fuel content showed increases in a 0.078-inch-diameter spot of up to 25 percent. Assuming such a spot to exist, it was calculated that an increase in the surface heat flux of eight percent resulted, hence the local fuel content factor of 1.08.

Calculations and measurements of fuel plate deflection caused by thermal and pressure induced stresses indicate that the resultant channel closures occur locally and are very small. Therefore, factors for operating plate deflection effects were not included.

3.2 Statistical Analysis

In applying the error propagation method, it is necessary to assign relative deviations (for two standard deviations) to each of the parameters. The values used are cited in Table V-3-4. The bases for these values are the same as those for the maximizing factors.

TABLE V-3-3

SUMMARY OF WALL TEMPERATURE MAXIMIZING FACTORS

Heat Flux	
Power measurement and control	1.05
Fuel content, local, including fuel per plate	1.08
Fuel core area	1.04
ATRC to ATR correspondence	1.05
Fin effect	0.98
Fraction of reactor power generated in fuel	<u>0.93</u>
	1.13
Hot Channel	
Coolant channel geometry	1.084
Fuel content, hot track (including fuel core area)	1.03
Flow distribution	1.05
Power measurement and control	1.05
Fuel per plate	1.01
Transverse velocity	1.00
ATRC to ATR correspondence	1.05
Fin effect	0.98
Fraction of reactor power absorbed by channel coolant	<u>0.97</u>
	1.24
Hot Spot	
Heat transfer coefficient	<u>1.25</u>
	1.25
Other Data	
Minimum velocity	44.5 fps
Axial maximum to average ratio	1.43
Radial maximum to average ratio	2.66

FIG. V-3-1 AXIAL TEMPERATURE PROFILES ALONG THE HOT FUEL PLATE AND COOLANT CHANNEL.

FIG. V-3-2 AXIAL TEMPERATURE PROFILES ALONG THE HOT FUEL PLATE AND COOLANT CHANNEL.

FIG. V-3-3 AXIAL TEMPERATURE PROFILES ALONG THE HOT FUEL PLATE AND COOLANT CHANNEL.

FIG. V-3-4 AXIAL TEMPERATURE PROFILES ALONG THE HOT FUEL PLATE AND COOLANT CHANNEL.

TABLE V-3-4

ERROR PROPAGATION DATA

Factor	Relative Deviation
Reactor power, 250 MW	0.05
Velocity, 47.8 fps	0.15
Fractional heat added, 0.68	0.10
Heat transfer area, 2.30 ft ²	0.10
Channel width, 3.752 in.	0.01
Channel thickness, 0.078 in.	0.10
Radial peaking factor $(1 + Z_1)$ 2.66	0.05
Average fuel plate content $(1 + Z_2)$ 1.00	0.10
Average velocity deviation $(1 + Z_3)$ 1.00	0.05
Axial peaking factor $(1 + Z_4)$ 1.348 (4 in. below centerline)	0.10
Local fuel plate content $(1 + Z_5)$ 1.00	0.10
Local velocity deviation $(1 + Z_6)$ 1.00	0.10
$f_1 = (1 + Z_1) (1 + Z_2) = 2.66$	
$f_2 = 1 + Z_3 = 1.00$	
$f_3 =$ fuel plate heat generation, 0.93	0.05
$f_4 = (1 + Z_1) (1 + Z_2) (1 + Z_4) (1 + Z_5) = 3.58$	
$f_5 =$ heat transfer coefficient multiplier = 1.00	0.20
$f_6 = 1 + Z_6 = 1.00$	
$f_7 =$ fin effect = 0.98	0.10
$f_8 =$ channel coolant heat transport = 0.97	0.05

The basis for the estimate of the wall temperature standard deviation is an error propagation of a correlation (determined by regression analysis) relating the wall temperature to the bulk water temperature, the heat flux, and those parameters related to the heat transfer coefficient (as determined for example by Dittus and Boelter or Colburn). The particular correlation for the heat transfer coefficient used in Reference 22 was the Modified Colburn Equation; in the range of the ATR operating conditions this correlation and the Dittus-Boelter equation result in about the same wall temperature (within about five percent). Both correlations have the following form with the only exception being that the physical properties are evaluated at the local film temperature in the Modified Colburn Equation and at the local bulk water temperature in the Dittus-Boelter Equation:

$$\frac{hDe}{k} = 0.023 Re^{0.8} Pr^{\frac{1}{3}}$$

where

h = heat transfer coefficient, Btu/hr-ft² - °F

De = hydraulic diameter, ft

k = thermal conductivity, Btu/ft-hr - °F

Re = Reynolds number, dimensionless

Pr = Prandtl modulus, dimensionless.

The correlation determined by regression analysis for the wall temperature is:

$$T_w = 46.7 + 0.87 T_b + 16.99 \times 10^{-4} K^*$$

where

T_w = wall temperature, °F

T_b = local bulk water temperature, °F

$K^* = q/A \times 1/B$

q/A = local heat flux, Btu/hr-ft²

$B = V^{0.8}/de^{0.2}$

V = velocity, fps

de = hydraulic diameter, in.

An error propagation of the above correlation resulted in the equation developed in Reference 22 to determine the surface (wall) temperature standard deviation.

The standard deviation computed was 55°F at 250 MW and maximum lobe power split. The maximum nominal wall temperature was calculated to be

350°F. The wall temperature at which incipient nucleate boiling occurs was calculated to be 462°F. Therefore, the nominal wall temperature is lower than two standard deviations compared to the nucleate boiling point. Details of the calculations are shown in Appendix F.

3.3 Thermal Burnout Analyses

Thermal burnout was evaluated using two different methods with the lower value of the two methods used in establishing the various instrument setpoints.

The first method resulted from observations made during the experimental measurements wherein burnout was obtained whenever the temperature of the bulk water exiting from the channel reached the saturation temperature at the local static pressure.

Local boiling burnout for conditions similar to those specified for the ATR have been measured and correlated by L. Bernath [23]. The correlation constituted the basis for the second method.

The hot track concept together with the hot channel factors, heat flux factors, and other maximizing factors were incorporated into the calculation of the burnout heat flux and operating maximum heat flux. The trip setpoint was adjusted so the ratio of the burnout heat flux to operating maximum heat flux was not below 1.0, thereby providing a reasonable probability that thermal burnout will not occur if all the maximizing conditions are realized simultaneously. Table V-3-5 summarizes the thermal burnout calculations.

An error propagation analysis of the bulk water rise in the hot channel was performed to assess the risk of obtaining thermal burnout if the reactor were inadvertently operated near any one of the several setpoints. The results of these calculations are presented in Table V-3-6. The basic criteria for this analysis was the assumption that thermal burnout occurs whenever the hot track, bulk coolant temperature reaches the saturation temperature corresponding to the local static pressure.

In order to perform an error propagation determining the standard deviation of the bulk water temperature and relating this to the local saturation temperature, a correlation, determined by a polynomial fit of the thermodynamic properties of saturated steam, was developed:

$$T_{\text{sat}} = 132.5 + 0.3852P - (0.3806 \times 10^{-3}) P^2$$

where:

T_{sat} = saturation temperature, °C

P = saturation pressure, psia.

TABLE V-3-5

THERMAL BURNOUT SUMMARY

Reactor Power Transients	
Ratio of burnout heat flux to operating maximum heat flux at 250 MW	1.6
Ratio of burnout power level to operating power level	1.25
Burnout margin at the trip level	1.04
Reactor Pressure Transients	
Ratio of burnout pressure to nominal at coolant channel inlet	0.70
Reactor Flow Transients	
Ratio of burnout velocity to nominal	0.82

TABLE V-3-6

RESULTS OF NOMINAL HOT CHANNEL ERROR PROPAGATION ANALYSES

	Bulk Water Temperature Standard Deviation (°F)	Hot Track Number of Standard Deviations
Reactor Power Transients		
Reactor power = 250 MW	20	5
Reactor power = 1.25 x 250 = 313 MW	25	3
Nominal flow and pressure		
Reactor Pressure Transients		
Inlet pressure = 0.70 normal	20	3
Reactor power = 250 MW		
Nominal flow		
Reactor Coolant Flow Transients		
Coolant velocity = 0.82 normal	25	3
Reactor power = 250 MW		
Nominal pressure		

Standard thermal-hydraulic equations were written to establish the bulk water temperature and local static pressure at the exit of the channel. The standard deviation of the bulk water temperature is the square root of the following function:

$$V(\phi) = \sum (\partial\phi/\partial X)^2 V(X)$$

where

$V(\phi)$ = the variance of ϕ

$\partial\phi/\partial X$ = the partial derivative of each of the affected parameters
and

$V(X)$ = variance of each parameter.

The variances used in the error propagation analysis are shown in Table V-3-4.

3.4 Low Flow and Decay Heat Removal

Decay heat removal is accomplished by at least one of the two emergency coolant pumps in the event the primary coolant pumps fail. The emergency coolant pump provides 10 percent of normal flow. The primary coolant coast-down rate compared to power decay rate following the scram shows adequate

flow is always available to remove decay heat (see loss of coolant flow accident -- Section XVI) during this period. The flow requirements to prevent burnout during emergency flow conditions are based on isolated hot channel conditions and the following criteria:

- (1) Heat flux equal to 60 percent of the burnout heat flux predicted by Bernath's equation
- (2) No net or local boiling
- (3) Reynolds number of 5000
- (4) No flow instability due to thermal buoyancy (the ratio of Grashoff number to Von Karman number must be equal to or less than 1).

The analytical results show that:

- (1) Degassing and depressurizing may be accomplished 17 minutes after loss of primary coolant pump circulation with three percent of normal flow. The governing criteria in this case are burnout and possible flow instability due to buoyancy.
- (2) Flux mapping runs at 2.5 MWt are less restrictive than item (1).
- (3) The elements left in the core one hour after shutdown are adequately cooled by flow from one decay heat coolant pump if the bypass flow is no more than obtained when five elements are removed from the core. Tests have been specified to determine if an element removed at this time is adequately cooled during the flow reversal that could occur as the element is moved from the core to the storage racks. However, data [24] indicate that ATR elements can be adequately cooled in a completely restricted downcomer condition five hours after the reactor has been shutdown.

Natural circulation can occur within the ATR core with the fuel elements acting as a riser and the internals as downcomer. Calculations indicate that, once established, natural circulation will cool the core at atmospheric pressure 30 minutes after loss of power.

V-4. REFERENCES

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VI. PRIMARY COOLANT SYSTEM

- VI-1 System Arrangement and Control
- VI-2 Design of Principal Components
- VI-3 Primary Coolant Related Systems

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VI-1. SYSTEM ARRANGEMENT AND CONTROL

1. GENERAL

The reactor is cooled by circulating pressurized demineralized water in a closed primary loop. The primary loop pumps have a combined capability of 54,000 gpm while retaining an 18,000-gpm pump on standby. The expected coolant requirement at 250 MW(t) is 47,000 gpm, yielding 7000 gpm excess cooling capacity.

The coolant enters the reactor vessel through two 24-inch pipes at 125°F and 355 psig and leaves through four 18-inch outlet pipes at 167°F and 246 psig when the reactor is operating at a power level of 250 MW(t). After leaving the reactor, the coolant flows through a common 36-inch header to four parallel shell and tube heat exchangers, then is pumped back to the reactor by three of the four 18,000-gpm primary coolant pumps which discharge into a common 36-inch header. The primary pumps can be individually isolated from the system whereas the primary heat exchangers cannot be isolated on the primary coolant (tube) side.

Auxiliary systems are provided for gland sealing, coolant makeup, flushing, degassing, demineralization, pH control, and emergency coolant supply. The primary coolant flow paths are shown in Figure VI-1-1. The primary coolant operating conditions are presented in Table VI-1-1.

2. METHODS OF CONTROL

2.1 Flow Control

The primary coolant flow is adjusted by a motor-operated butterfly-type valve which is positioned before reactor start-up. Minor adjustments to the butterfly valve may be required after reactor start-up and during operation. A mechanical stop is provided to prevent full stoppage of coolant flow and to prevent primary coolant pumps from operating at shutoff head. Flow adjustment is based on maintaining 100 psi across each quadrant.

Primary coolant flow is measured by orifice-type flow meters located in each of the four reactor outlet lines.

Reactor core quadrant flows and/or differential pressure across the flow dividers can be balanced, if required, by the installation of flow restricting orifice plates in each of the four reactor outlet lines.

2.2 Temperature Control

Reactor inlet temperature is controlled by a three-element system that varies the secondary coolant flow to the primary heat exchangers.

2.3 Pressure Control

The reactor inlet coolant pressure is regulated by controlling the inventory of coolant. The pressurizing and gland seal pumps discharge into the system and the pressure control valve bleeds from the system to maintain constant pressure.

FIG. VI-1-1 PRIMARY COOLANT SYSTEM FLOW SHEET

TABLE VI-1-1

PRIMARY COOLANT SYSTEM OPERATING CONDITIONS

Reactor power	250 MW(t)
Reactor inlet temperature (nominal)	125°F 130°F (design)
Reactor outlet temperature (nominal)	167°F 169°F (design)
Core inlet pressure	347 psig
Core pressure drop	100 psi
Fuel element flow rate	27,400 gpm
Total reactor (nominal) flow rate	47,000 gpm
Number of primary coolant pumps	4 (one spare)
Capacity of each primary coolant pump	18,000 gpm
Equipment design pressure	390 psig
Equipment design temperature	240°F
Piping design pressure and temperature:	
(1) From primary pump discharge to motor operated butterfly valve	485 psig-150°F
(2) From motor operated butterfly valve to primary pump suction	390 psig-240°F

A surge tank air cushion is provided to damp minor pressure fluctuations. The total volume of the surge tank is 1000 cubic feet of which approximately 50 cubic feet is air when the normal primary system operating pressures and flow are maintained.

2.4 Coolant Quality Control

The primary coolant purity is maintained by a primary coolant bypass demineralizer system and a degassing system to clean up and degas the coolant.

2.5 Emergency Cooling

Two 4700-gpm emergency coolant pumps (one a spare) operate in the closed primary loop. Their capacity is based on flow requirements to prevent reactor core steam blanketing during low-power flux runs and reactor flow requirements after loss of electrical power to all primary coolant pumps.

To assure the required flow after primary coolant pump coastdown without reliance on emergency pump start-up, one emergency coolant pump, driven by an

ac motor supplied with diesel-commercial power, operates continuously during reactor operation. The pump normally discharges 250 gpm through a recirculation line to the primary pump suction header. On loss of primary coolant pressure, the check valve on the emergency coolant pump discharge (normally closed by primary pump pressure) opens, enabling the emergency coolant pump to circulate 4500 gpm in the primary coolant loop. The recirculation flow falls to 200 gpm when this occurs.

The standby emergency coolant pump, connected in identical fashion, is driven by a dc motor supplied directly from a battery. This battery is float-charged by a diesel-power supplied, motor-generator set.

The battery power source for the dc motor-driven, emergency coolant pump is sized for 30 minutes of operation without recharging. Thirty minutes is sufficient time to start the standby diesel engine and restore diesel power to the battery-charging motor-generator set if the operating diesel engine simultaneously fails with commercial power.

The dc driven emergency coolant pump is automatically started and reaches full speed within seven seconds. Automatic starting is provided to backup the operating ac driven emergency coolant pump in the event of any of the following occurrences which also scram the reactor:

- (1) Mechanical or electrical failure of the ac driven pump, sensed by means of low flow switches in the recirculation line
- (2) Loss of diesel power sensed by low phase voltage switches
- (3) Loss of all primary pumps sensed by low flow switches in primary coolant line
- (4) Loss of commercial power (indirectly by loss of all primary pumps).

Should a scram occur for reasons other than loss of either or both diesel and commercial power, the dc pump is not started because the primary pumps and the ac emergency coolant pump remain in operation.

One emergency coolant pump is adequate for removal of reactor heat during low-power flux mapping runs. Both pumps are operated during the low-power runs so that loss of a single electrical system or pump will not result in a complete loss of flow. The coolant flow to the reactor vessel is 5400 gpm with both emergency coolant pumps in operation. A scram is initiated on low flow or loss of either pump.

VI-2. DESIGN OF PRINCIPAL COMPONENTS

1. GENERAL

The equipment and piping used in the reactor primary coolant system are designed, fabricated, and erected in accordance with or exceeding applicable codes and utilize designs proved reliable. The quality of this work was assured through inspection performed by qualified inspectors with experience in the various classes of equipment and materials used.

The materials in contact with the primary coolant are type 304 stainless steel with the additional requirement that type 304L stainless steel be used for piping in areas that are inaccessible for maintenance or inspection. Those portions of the heat exchanger in contact with both primary and secondary coolant are Inconel. Inconel is used in order to resist possible chloride stress corrosion from the secondary side. One of the four heat exchangers contains several Type 304 stainless steel tubes for materials testing purposes.

All mechanical seals on primary coolant system pumps, and all reactor penetration seals are supplied with high-pressure, demineralized gland seal water to the outside ends of the seals to prevent the leakage of primary coolant from the sytem.

2. PRIMARY HEAT EXCHANGERS

2.1 Arrangement

The four primary heat exchangers, arranged in parallel, are located in the northwest side of the second basement and extend upward into the first basement. Hatch covers in the main floor permit servicing.

2.2 Design Characteristics

Fluid	<u>Tube Side</u>	<u>Shell Side</u>
	Primary coolant water	Secondary coolant water
Fluid treatment	Demineralization, degassing, and pH adjustment	6.3 pH, chlorine, sulphuric acid, and phosphate addition
Total quantity	2.35×10^7 lb/hr	1.91×10^7 lb/hr
Temperature in (°F)	167	85
Temperature out (°F)	124	138
Total heat transferred	9.17×10^8 Btu/hr	
Pressure drop	13 psi	22 psi

2.3 Materials of Construction

Tubes	Inconel
Tube type	U-bend
Tube size	3/4-inch OD x 18 Bwg
Tube Sheets	Inconel clad carbon steel
Tube joints	Welded
Shells	Carbon steel
Tube support plates and baffles	Carbon Steel
Channels and channel covers	Type 304 stainless steel clad carbon steel

2.4 Codes and Tests

The primary heat exchangers are designed and constructed in accordance with the ASME Unfired Pressure Vessel Code, Section VIII, 1959, latest addenda; and General Requirement for Nuclear Vessels, Special Ruling Case 1270 N-5 and were tested and inspected as follows:

- (1) Shell and tube side of each heat exchanger were hydrostatically tested in accordance with paragraph UG-99 of the ASME Unfired Pressure Vessel Code, 1959.
- (2) All shell and channel pressure welds were fully radiographed in accordance with paragraph UW-51 and Code Case 1270 N of the ASME Unfired Pressure Vessel Code, 1959.
- (3) All tubes were completely liquid-penetrant tested prior to bending, and after bending the bent area was liquid-penetrant tested in accordance with the Commercial Dye Penetrant Test Procedure of the Huntington Alloy Products Division of the International Nickel Company, Inc.
- (4) All welded joints, tube-to-tube sheets, were liquid-penetrant tested in accordance with manufacturer's standard.
- (5) Clad-tube sheet plates and all tubes were examined by ultrasonic test (Immersion Technique) in accordance with the applicable Nondestructive Testing Method described in MIL-STD-271A.

3. PRIMARY COOLANT PUMPS

3.1 Arrangement

The four primary coolant pumps, arranged in parallel, are located in the first basement, south of the heat exchangers. Hatch covers in the main floor permit servicing individual cubicles.

3.2 Design Characteristics

Quantity	4 (one spare)
Fluid pumped	Primary Coolant Water
Pump type	Horizontal, double suction, volute centrifugal
Capacity, each	18,000 gpm
TDH	400 ft
Normal suction pressure	215 psig
Normal suction temperature (°F)	125
Speed	1750 rpm
Brake horsepower	2190 hp
Motor horsepower	2000 hp
Moment of inertia of rotating pump and motor rotor	2000 lb ft ²

3.3 Materials of Construction

Casing	Austenitic stainless steel
Impellers	Austenitic stainless steel
Casing rings	12 percent chrome steel
Impeller rings	Austenitic stainless steel
Shaft	SAE 4140 steel
Shaft sleeves	Austenitic stainless steel
Shaft seals	Mechanical face type
Bearings	Ball or roller

3.4 Codes and Tests

The primary coolant pumps were designed and constructed in accordance with the Centrifugal Pumps Section of the Standards of the Hydraulic Institute, 10th edition, and tested and inspected as follows:

(1) All four pumps were shop tested with cold water at rated speed from shutoff head to 120 percent of design capacity. The following data were recorded:

(a) Head versus capacity

(b) Overall efficiency versus capacity

(c) kW input versus capacity

(d) Results of seal leakage tests

(2) Primary coolant system head curves with one pump, two pump, three pump, and four pump operation are shown in Figure VI-2-1. Primary coolant system hydrodynamic curves which show hydraulic and pressure gradients of the system with 47,000 gpm and 54,000 gpm are presented in Figure VI-2-2.

FIG. VI-2-1 PRIMARY COOLANT SYSTEM PUMP HEAD CURVES.

(3) All pump casings were hydrostatically tested at 585 psig.

(4) All pump tests were in accordance with ASME Power Test Code for Centrifugal Pumps, PTC 8.1, 1954 edition.

4. EMERGENCY COOLANT PUMPS

4.1 Arrangement

The two emergency coolant pumps, arranged in parallel, are located west of the primary coolant pumps in the first basement. A hatch cover in the main floor permits servicing.

FIG. VI-2-2 PRIMARY COOLANT SYSTEM HYDRODYNAMICS.

4.2 Design Characteristics

Quantity	2 (one spare)
Fluid pumped	Primary coolant water
Pump type	Horizontal, double suction, volute centrifugal
Capacity, each	4700 gpm
TDH	22 ft
Normal suction pressure	215 psig
Emergency operation suction pressure	
At start	200 psig
30 minutes after start	Atmospheric with flooded suction
Normal suction temperature (°F)	125

Speed	690 rpm
Brake horsepower	39 hp
Motor horsepower	40 hp
Motor types	One ac and one dc

4.3 Materials of Construction

The materials of construction are identical with those for the primary coolant pumps.

4.4 Codes and Tests

The emergency coolant pumps were designed and constructed in accordance with the Centrifugal Pumps Section of the Standards of the Hydraulic Institute, 10th edition, and were tested and inspected as follows:

(1) Both pumps were shop tested with cold water at rated speed from shut-off head to 120 percent of design capacity. The following data were recorded:

- (a) Head versus capacity
- (b) Overall efficiency versus capacity
- (c) kW input versus capacity
- (d) Results of seal leakage tests
- (e) Pump vibration amplitude

Amplitude, peak to peak

Amplitude, maximum displacement from normal

(2) Both pumps also were shop tested at a capacity of 250 gpm for six hours duration to check performance at the recirculation operating condition. The following data were recorded every five minutes:

- (a) Head
- (b) kW input
- (c) Pump efficiency
- (d) Seal leakage in gpm
- (e) Temperature rise of water through the pump
- (f) Pump vibration

Amplitude, peak to peak

Amplitude, maximum displacement from normal

(3) Performance curves for both pumps are shown on Figures VI-2-3 and VI-2-4.

FIG. VI-2-3 EMERGENCY COOLANT PUMP (670 M-10) PERFORMANCE CURVE.

FIG. VI-2-4 EMERGENCY COOLANT PUMP (670 M-11) PERFORMANCE CURVE.

(4) Both pump casings were hydrostatically tested at 585 psig.

(5) All pump tests were in accordance with ASME Power Test Code for Centrifugal Pumps, PTC 8.1, 1954 edition.

5. PRIMARY PRESSURIZING PUMPS

5.1 Arrangement

The two pressurizing pumps, arranged in parallel, are located west of the heat exchangers in the second basement.

5.2 Design Characteristics

Quantity	2 (one spare)
Fluid pumped	Demineralized water
Pump type	Horizontal, volute centrifugal
Capacity, each	300 gpm
TDH	695 ft
NPSH	10 ft
Suction temperature (°F)	169
Speed	3550 rpm
Brake horsepower	89 hp
Motor horsepower	100 hp

5.3 Materials of Construction

The materials of construction are identical with those for the primary coolant pumps.

5.4 Codes and Tests

The primary pressurizing pumps were designed and constructed in accordance with the Centrifugal Pumps Section of the Standards of the Hydraulic Institute, 10th edition, and were tested and inspected as follows:

(1) Both pumps were shop tested with cold water at rated speed from shut-off head to 120 percent of design capacity. The following data were recorded:

- (a) Head versus capacity
- (b) Overall efficiency versus capacity
- (c) kW input versus capacity

(2) Performance curves for both pumps are shown in Figures VI-2-5 and VI-2-6.

(3) All pump casings were hydrostatically tested at 585 psig.

(4) All pump tests were in accordance with ASME Power Test Code for Centrifugal Pumps, PTC 8.1, 1954 edition.

6. SURGE TANK

6.1 Design Characteristics

Type	Vertical
Inside diameter	8 ft
Height, tangent to tangent	18 ft

Capacity	1000 cu ft
----------	------------

Heads	Elliptical
-------	------------

6.2 Materials of Construction

Shell and heads	Type 304 stainless steel clad carbon steel
-----------------	---

Nozzles	Type 304 stainless steel clad carbon steel
---------	---

6.3 Codes and Tests

The surge tank was designed and constructed in accordance with the ASME Unfired Pressure Vessel Code, Section VIII, 1959, latest addenda; and General Requirements for Nuclear Vessels, Special Ruling Case 1270 N-5 and was tested and inspected as follows:

(1) The tank was hydrostatically tested in accordance with paragraph UG-99 of the ASME Unfired Pressure Vessel Code, 1959.

(2) All welds were fully radiographed in accordance with paragraph UW-51 and Code Case 1270 N of the ASME Unfired Pressure Vessel Code, 1959.

7. DEGASSING TANK

7.1 Design Characteristics

Type	Vertical - spray
------	------------------

Inside diameter	5 ft
-----------------	------

Height, overall	28 ft
-----------------	-------

Heads	Elliptical
-------	------------

Design pressure	65 psia and 3 psia
-----------------	--------------------

Water flow rate	600 gpm
-----------------	---------

Air flow rate	100 scfm
---------------	----------

7.2 Materials of Construction

Shell and heads	Type 304 stainless steel
-----------------	--------------------------

Nozzles and internals	Type 304 stainless steel
-----------------------	--------------------------

7.3 Codes and Tests

The degassing tank was designed and constructed in accordance with the ASME Unfired Pressure Vessel Code, Section VIII, 1959 edition, and the General

Requirement for Nuclear Vessels, Special Ruling Case 1270 N-2, and inspected and tested as follows:

(1) The vessel was inspected and stamped in accordance with ASME Code for Unfired Pressure Vessels, Section VIII, 1959 edition, paragraphs UG-90 through UG-120, UW-46 through UW-52, and UCL-50 through UCL-55 and the General Requirements for Nuclear Vessels, Special Ruling Case 1270 N-2.

(2) The completed tank and internals were hydrostatically tested at 1.5 times the maximum design pressure, and vacuum tested at 3 psia.

8. PIPING

8.1 General Arrangement -- Materials of Construction -- Codes

All piping materials are Type 304 stainless steel except that Type 304L is used in inaccessible areas. Buried and embedded piping is 304L either seamless per ASTM A376-60T, or welded per ASTM A358-60T as stated in ASA B31.1-1955 Code for Pressure Piping Case N-7. 304L stainless steel piping met the chemical requirements of ASTM A-312-61T. Dimensions for nominal wall pipe are in accordance with ASA B36.19-1957.

Instrument tubing is fully annealed stainless steel per ASTM A269-61. The tubing is 1/2-inch OD or smaller with 0.035-inch wall thickness suitable for bending and flaring. Instrument taps on the primary coolant system are 1-inch pipe.

8.2 Welding

Stainless steel pipe welding was performed by the tungsten inert gas process, or by the combined tungsten inert gas and shielded metal arc process. All welding operators were qualified under the applicable code requirements, and all welds are marked with the welding operator's identification. All longitudinal welds in all pipes, and welded joints on pipes four-inch diameter and larger, to and including the first isolation valve on branch piping for auxiliary systems, are radiographed. Socket welds, fillet welds, and hanger attachment welds were not radiographed. All socket, fillet, and groove welds were inspected for surface defects by a liquid penetrant method, and all pressure welds which could not be satisfactorily radiographed were inspected by the liquid penetrant method after completion of each 1/2 inch of weld thickness. All repair welding of defects was reinspected.

8.3 Primary Coolant System Pipe Stresses

The primary coolant system piping allowable stresses range from 25,865 psi for Type 304L material to 28,125 psi for Type 304 material at 170°F working temperature. The following tabulation lists the maximum calculated piping stresses for the indicated portion of the primary loop as shown on Figures VI-2-7 and VI-2-8.

<u>Primary Coolant System Section</u>	<u>Maximum Stress, psi</u>
Reactor vessel outlets to heat exchanger inlet header (Type 304 and 304L)	9,223
Heat exchanger inlet header to heat exchanger inlets	15,620
Heat Exchanger outlets to primary pump suction	9,400
Primary pump discharges to reactor inlet header	11,250
Reactor inlet header to reactor vessel inlets (Type 304 and 304L)	12,570
Heat exchanger outlet to surge tank	5,500

The stress values have taken into account temperature differentials due to reactor power shifts which may be required by a specific complement of experiments. Calculations indicate that the most severe case of temperature unbalance will produce 155°F coolant from one quadrant, when an adjacent quadrant produces 170°F coolant. The maximum stress for this case is 9223 psi, as listed above.

Since the maximum allowable stresses include a factor of safety of 4, and the maximum primary coolant system pipe stress is 15,620 psi, the minimum overall safety factor is 7.

8.4 Flexibility and Methods of Support

The piping system and connected equipment is supported, restrained, and guided so as to limit pipe stresses to the values listed. Adjustable constant support-type spring hangers will be used to carry primary system piping loads.

A complete stress analysis was made on the primary coolant system piping, and the results are shown on Figures VI-2-7 and -8. A complete flexibility analysis and method of pipe support are available on ATR drawings.

8.5 Vibration Analysis

Field tests of all rotating equipment and associated systems will be made to determine if any areas of excessive vibration exist.

9. VALVES

All valves in contact with the primary coolant water are 300-pound Standard ASA, Type 304 stainless steel, forged, cast or fabricated, per ASTM A182-61T or ASTM A296-60T, Grade CF-8, except the primary relief valves which are carbon steel of ASTM-A216 60T, Grade WCB. All primary system valves were hydrostatically tested at 585 psig, and all valves four inches and larger up to and including the first isolation valve on branch piping for auxiliary systems were radiographed.

PRIMARY COOLANT SYSTEM

FIG. VI-2-7 PRIMARY SYSTEM PIPE STRESS ANALYSIS.

VI-3. PRIMARY COOLANT RELATED SYSTEMS

1. SURGE TANK AND PRESSURIZING SYSTEM

Compressed air from the 250-psig system is used to supply the surge tank. About 50 cubic feet of air at 215 psig is normally contained in the 1000-cubic-foot surge tank which is connected to the primary coolant loop near the primary heat exchangers outlet. With the primary coolant pumps operating, the reactor inlet pressure is established at 355 psig. Rapid and short duration pressure and temperature induced transients are damped by expansion or compression of the surge tank air inventory. Slower and long duration pressure variations are damped by controlling the system inventory of coolant. The reactor inlet pressure is controlled by automatic adjustment of the system outflow to the degassing tank through a pressure control valve to balance the preset flow from pressurizing pumps and in-leakage from various seals. The pressurizing system instrumentation is described in Section VII.

Normally, the degassing flow rate is preset to establish a flow rate of 50 to 300 gpm. The expected in-leakage of gland seal water is about 70 gpm from reactor penetration seals and pump shaft seals.

Compressed air is admitted periodically to the surge tank to restore air inventory lost through solution in the primary coolant. All admission or discharge of air is under manual control. Double control valves, with an atmospheric bleed valve between the valves, and a check valve are used for admitting compressed air to the surge tank. Double control valves are also used for air discharge. In addition, a surge tank low-level switch will automatically shut the air admission valves. This prevents an excessive air inventory which could overflow from the surge tank into the primary circulating system in the event that coolant volume is reduced by contraction, leakage, and/or depressurization. After reactor shutdown, compressed air is manually discharged from the surge tank to the degassing tank when the system pressure is reduced to atmospheric.

2. PRESSURE RELIEF SYSTEM

Two relief valves mounted on the reactor vessel, eight-inch overflow line protect the reactor from overpressurization. Each valve is designed to discharge 600 gpm at 379 psig. A safety relief valve mounted at each pump set at 290 psig to relieve 300 gpm protect the primary pressurizing pumps from overpressurization when these pumps are used for primary system cleanup in conjunction with the bypass demineralizers. No safety relief valves are required for the emergency coolant pumps because they are not capable of developing a discharge pressure greater than the primary coolant system design pressure.

3. PRIMARY COOLANT pH CONTROL SYSTEM

The primary coolant pH is controlled between 5.0 and 6.0 in order to minimize corrosion of the aluminum-clad fuel elements. Reactor coolant water

is sampled at the reactor inlet by a pH controller which adjusts the rate of nitric acid injection into the primary coolant system. The acid is injected upstream of the heat exchangers to obtain adequate mixing before entering the reactor vessel. The pH control system instrumentation is described in Section VII.

The injection system utilizes two dc motor-driven positive displacement pumps (one a spare) constructed of corrosion resistant material, each having a capacity of 5.4 gph at 600-psig discharge pressure. The pump speed, and thus delivery rate, is adjusted by the pH controller through a statatrol rectifier which provides power to the dc motor drive in proportion to the system acid requirements. The range of pH control of this acid injection system is 3.5 ± 0.2 to 6.0 ± 0.2 .

4. BYPASS DEMINERALIZER SYSTEM

4.1 System Description

Normally, 100 gpm of reactor primary coolant water is continuously bypassed through one of a dual train demineralizer system and returned to the suction side of the primary coolant pumps. Each train consists of a carbon steel, rubber-lined cation unit and an anion unit. Both trains discharge through a common effluent filter. The demineralizer removes both dissolved and suspended corrosion products to produce water containing 1.0-ppm maximum ion concentration at the reactor vessel inlet. A flow sheet for the bypass demineralizer is shown in Figure VI-3-1.

The cation resin when spent will be too radioactive for economical regeneration; therefore, it is sluiced outside the building to a shielded tank truck for disposal at the NRTS burial grounds. A replacement charge of fresh cation resin is stored in the resin fill tank.

One anion resin regeneration tank and one regenerated anion resin storage tank serve both anion units with all resin transferred by water and air sluicing. Caustic is stored in a portable tank and is used for anion resin regeneration.

4.2 System Capacity

The water purification system has the capacity to demineralize up to 600 gpm of primary coolant water during flushing operations or to clean up a release of fission and radioactive corrosion products during reactor operation.

4.3 Materials of Construction

Natural rubber linings that retain their useful properties up to 10^8 rad integrated beta-gamma dose are used for the cation and anion units. Calculations indicate that these linings will adequately protect the carbon steel tanks from contact with primary coolant and contaminated resin for approximately ten years at which time the linings would be replaced. Experience at MTR and ETR indicates that this life expectancy is realistic.

All demineralizer piping and valves are designed and fabricated of the same materials and to the same pressure standards as the primary coolant system.

FIG. VI-3-1 BYPASS DEMINERALIZER FLOW SHEET.

Cation and Anion Unit Design Characteristics

Diameter	60 inches
Straight shell height	72 inches
Shell and head material	Carbon steel
Lining	3/16-inch natural rubber
Internals	Durimet 20
Design pressure	400 psig

4.4 Codes and Tests

The bypass demineralizer cation and anion tanks are designed and constructed in accordance with the ASME Unfired Pressure Vessel Code, Section VIII, 1959 and were tested and inspected as follows:

- (1) Tanks were hydrostatically tested at 600 psig prior to release for rubber lining. Vessels are code stamped.
- (2) All rubber linings are fabricated from sheet-form natural rubber, 3/16 inch thick, and vulcanized in place. Rubber linings extend from the tank through openings in an uninterrupted continuous manner and up over flange faces to the drilling circle.
- (3) Completed rubber linings were subjected to a megger test. Rubber-linings were continuous, smooth, and without voids.

4.5 Operating Limitations

The resin beds are protected from high temperature water by primary system temperature control.

5. DEGASSING SYSTEM

5.1 System Description

A degassing tank serves to remove gases so that bubbles will not form in any part of the primary system. The gas concentration in the coolant is maintained in dynamic equilibrium by continuously spraying a preset quantity of reactor coolant, at reactor exit temperature and pressure, into the tank which is maintained at slightly less than atmospheric pressure.

5.2 Gas Removal Design and Capacity

The system design is based upon the following calculated gas production rates:

	<u>CC (STP) /Minute</u>
Radiolytic decomposition	1800
H ₂ from corrosion	100

	<u>CC (STP) /Minute</u>
Air from the surge tank	4870
Air in seal and makeup water	3410
Krypton and xenon (0.5 percent gas in 1 fuel plate/cycle)	Kr- 0.441×10^{-5} Xe- 2.38×10^{-5}

Degassing flow rate varies between 50 to 300 gpm during normal operation.

With the above gas production rates and available degassing flow rate, the maximum equilibrium gas concentrations in the primary coolant will be as follows:

$$\begin{aligned} \text{H}_2 &- 1.15 \text{ cc/liter} \\ \text{O}_2 &- 5.62 \text{ cc/liter} \\ \text{N}_2 &- 11.7 \text{ cc/liter.} \end{aligned}$$

These maximum gas concentrations correspond to 14.7 percent of saturation at the pressure and temperature in the main coolant pump suction, which are the conditions of minimum gas solubility. The equilibrium concentration of gaseous fission product activity is $0.024 \mu\text{Ci/cc}$ based on the above release rate during normal operation.

The water level in the degassing tank is automatically maintained by a split-range, liquid-level controller which operates coolant makeup and blowdown valves. An overflow line is provided to handle abnormally large blowdown volumes. Blowdown and overflow discharge to the warm waste tank. The accumulation of gases is continually removed by the action of a supplementary fan which induces the flow of 100 cfm of sweep air through the degassing tank and exhausts it to the ventilation stack. This flow results in a H_2 concentration of 0.07 percent compared to the lower explosion limit of 4 percent. Furthermore, since the degassing tank has an air volume of 300 cubic feet, it would require air stagnation for 2.8 hours for the full degassing rate to produce a 4 percent H_2 concentration. An alarm is provided to annunciate loss of the degassing tank exhaust fan.

A vacuum-breaker type check valve is installed in the sweep air inlet line to the degassing tank to prevent blow-back of radioactive gas to the building in the event of exhaust fan stoppage, increased pressure in the degassing tank because of an operating mishap, or on discharge of the pressurizing pump relief valves.

5.3 Activity Monitoring

To reduce the concentration of noble gases following a release of fission products at approximately the same rate as the bypass demineralizer reduces the concentration of dissolved ionic products, flow rates through the two systems should be of the same order. Hence, the degassing system is designed for a maximum flow rate of 600 gpm. The degassing system, when operating at maximum capacity, has the capability of reducing the activity of the xenon and krypton in the coolant to $0.1 \mu\text{Ci/cc}$ within 24 hours if ten percent of the fuel

plate gaseous fission product inventory is released to the coolant. The degassing tank gas discharge is monitored for activity.

6. GLAND SEALING SYSTEM

6.1 System Description

The gland sealing system provides a high-pressure supply of demineralized water to the reactor vessel, primary system pump mechanical seals, and N-16 monitoring system. The seal water serves to prevent leakage of radioactive coolant to the building, lubrication for rotating seals, and prevents the deposition of crud on the sealing surfaces and in the reactor vessel bottom head. A flow sheet of the system is shown on Figure VI-3-2.

Sealing water is supplied to the reactor seals by the primary gland seal pumps and by the primary warm gland seal booster pumps.

The two 150 gpm primary gland seal pumps (one a spare) take their suction from the LDW system, which is supplied from the 200,000-gallon plant demineralized water storage tank, and discharge seal water at about 25 psi above reactor pressure.

6.2 System Supplied with Seal Water

Under normal conditions, the primary gland seal pumps supply demineralized water to the primary coolant system pump mechanical seals, top-head pressure tube seals, outer shim drum drive seals, N-16 monitoring system, and the reflector cylinder position indicators.

The two 150 gpm primary warm seal water booster pumps (one a spare) take their suction from the discharge of the bypass demineralizer and discharge demineralized primary coolant at about 25 psi above reactor pressure. Use of clean, primary coolant, rather than demineralized makeup water, for those seals which require an appreciable continuous inflow to prevent accumulation of crud, and where the minor radioactivity in the bypass demineralizer discharge can be tolerated, results in reduced requirements for makeup water and waste disposal. Under normal operating conditions, the primary warm gland seal booster pumps supply demineralized primary coolant to the bottom-head pressure tube seals, safety rod drive seals, neck shim drive housing seals, safety rod snubber tube flush, and reactor vessel bottom flushing. If for any reason the bypass demineralizer is out of service, the entire gland seal system is supplied by the primary gland seal pumps. This is accomplished by opening the gate valve, which interconnects the two sealing systems, and by stopping the warm gland seal booster pumps.

During reactor open-head operations, all reactor sealing is supplied by the primary gland seal pumps.

6.3 System Leak Detection

Rotameters equipped with high-flow alarms are provided for flow indication and will alarm if a seal leaks excessively.

7. DEMINERALIZED WATER MAKEUP SYSTEM

7.1 System Description

A 250-gpm capacity, dual train makeup demineralizer is installed in the existing demineralizer building MTR-608 to supplement the existing MTR-ETR demineralizer plant by providing a total capacity of 550 gpm for the entire facility. The demineralized water is stored in two existing 100,000-gallon stainless steel storage tanks. Two 500-gpm (one a spare) demineralized water transfer pumps supply the entire facility. In addition, two existing 1000-gpm flush pumps supply demineralized water for reactor flushing following each reactor cycle. Normally, only one reactor will be scheduled for shutdown at one time. A flow sheet of the system is shown on Figure VI-3-3.

7.2 System Uses

Demineralized water is supplied for the following services:

Reactor fill and makeup

Primary gland seal system

High-pressure demineralized water system fill

Experimental loop fill

Canal purge

Laboratories and other services.

8. REACTOR FLUSH SYSTEM

After a normal reactor shutdown at the end of the fuel cycle, primary coolant system flow, pressure, degassing rate, and, if necessary, the demineralizer flow rate are maintained at operating levels for up to four hours. This will reduce the activity of the coolant to an acceptable level for disposal to the facility leaching bed.

Following the period of demineralization and degassing, the reactor flush operation is started by reducing primary system pressure to 50 psig. This permits the entry of flush water at a rate of 2000 gpm from the plant demineralized water storage tanks. The primary coolant pumps are stopped, but the operating emergency coolant pump continues to run. The flush water is introduced at the suction side of the emergency coolant pumps and, after flowing through the reactor, is discharged from the system downstream of the primary heat exchangers to the area leaching pond for disposal. The flush piping is arranged to permit demineralizing up to 600 gpm flush water by the bypass demineralizer, if necessary, prior to disposal.

9. EMERGENCY WATER SUPPLY

9.1 Sources

Emergency supply of water to the reactor may be from either of the following sources:

- (1) The normal primary coolant demineralized water makeup and gland seal systems which are automatically actuated to supply maximum flow upon surge tank low level.
- (2) The fire-water supply system.

Minor leaks of coolant from seals, small leaks in piping, instrument tubes, etc, can be balanced by the pressurizing pump makeup system and the gland seal water system.

9.2 Sequence of Events Following Large Leakage

Should a coolant leak occur in excess of the normal makeup capacity, the surge tank water level will fall to the alarm switch level. The following automatic sequence of events occurs from the onset of low level:

- (1) The pressure control valve will close to prevent bleed from the primary system as a result of falling pressure which accompanies falling surge tank level.
- (2) The low-level switch prevents admission of pressurizing air to the surge tank, and signals the pressurizing pump flow control valve to the full-open position.
- (3) Additional water makeup is provided by the standby pressurizing pump which starts automatically on falling discharge header pressure.
- (4) The gland seal water pressure control valve opens wide on falling discharge header pressure, and the standby gland seal pump starts automatically.
- (5) Demineralized water supply for both pressurizing pumps is automatically provided by the degassing tank makeup valve which opens wide on falling degassing tank level.
- (6) The standby demineralized water transfer pump in Demineralizer Building MTR-608 starts automatically on falling discharge header pressure.

These actions supply a total of 1000 gpm to the primary coolant system of which 700 gpm is supplied by the pressurizing pumps to the suction header of the primary coolant pumps, and about 300 gpm is supplied by the gland seal pumps to the reactor vessel glands and bottom flushing connection. If this augmented quantity of makeup water successfully copes with the coolant leak, reactor pressure may be maintained for at least 100 minutes since at least one 100,000-gallon demineralized water storage tank is available. If both tanks are full, this time period is doubled.

Should the automatically augmented demineralized water makeup supply fail to cope with the system leak, the reactor water level will fall to the reactor outlet lines at which level suction to the emergency coolant pumps is lost. Calculations indicate that for ruptures up to 6.3 inches in diameter the emergency coolant pump will operate for at least 190 seconds before losing suction. At this time the minimum coolant flow required to prevent fuel melting is 2000 gpm.

9.3 Description of Fire-Water System

Emergency admission of water to the reactor core from the fire-water supply is initiated automatically when the water level in the reactor vessel falls to the center line of the outlet pipe lines. Two self-contained, level operated hydraulic control valves, LCV-1-7A and LCV-1-7B, each with a capacity of 2000 gpm, are installed in parallel to introduce fire water from an eight-inch line tied to yard fire protection loop to four two-inch connections on the reactor vessel bottom head within the area of the core tank. Depending on the location of the rupture, this system, in combination with the demineralizer makeup and gland seal systems, can supply up to 3000 gpm of water to the reactor vessel, 2300 gpm of which is applied for forced convection cooling of the core. This supply is adequate to remove reactor decay heat and prevent fuel element melting for ruptures up to eight inches diameter despite loss of suction to the emergency coolant pump.

Each two-inch line has a check valve close to the reactor bottom head to contain reactor water in the event of a rupture either in a two-inch line or the eight-inch, fire-water supply line. If a postulated rupture occurs in a two-inch, fire-water line between the reactor vessel and the check valve, the demineralized water makeup system, described earlier, can provide sufficient flow to maintain coolant inventory without losing suction to the emergency coolant pump.

The yard fire-water system is maintained at a minimum pressure of 55 psig by means of a 150,000-gallon, overhead, raw-water storage tank. The overhead tank is supplied by any of six existing MTR-ETR raw-water pumps ranging in capacity from 850 to 1300 gpm which take their suction from the three existing 500,000-gallon, ground-level, raw-water storage tanks. The ground-level tanks are replenished by three deepwell turbine pumps which have a total capacity of 9000 gpm. If the fire-water system pressure drops below 55 psig a 5100 gpm motor driven fire pump starts automatically and operates until fire system pressure is restored. Thus the 2000 gpm emergency fire-water flow to the reactor can be maintained indefinitely.

When the reactor decay heat requirements can be met by natural circulation, the fire water can be shut down, since either the demineralized water makeup, the gland seal, or the bottom flush water flows can provide the necessary 30 gpm of water to keep the core covered.

9.4 Analysis of Large Ruptures in the Primary System Handled By Emergency Water Supply

Ruptures located at an elevation above the reactor outlet lines do not affect operation of the emergency coolant pumps and do not require emergency makeup water to sustain their operation. For ruptures below the outlet lines, calculations indicate that when the reactor is at atmospheric pressure and the reactor water level falls to the level of the reactor outlet lines, the following leakage rates will occur:

<u>Item</u>	<u>Rupture Size (Inches)</u>	<u>Service</u>	<u>Location</u>	<u>Max. Leakage gpm</u>
1	1	Instruments	Lowest point in system	79
2	2	Fire water and bottom flushing	Reactor bottom head	205
3	5-1/2	Pressure tube	Reactor bottom head	1550
4	6	Surge tank connection	5 feet below bottom head	2070

By reference to the above tabulation and the previous discussion, certain ruptures below the elevation of the outlet lines can be adequately handled by the emergency water supply systems without losing suction to the emergency coolant pumps as follows:

(1) The demineralized water makeup via the pressurizing pumps alone (700 gpm) can balance the loss of water for ruptures up to 3.5 inches in diameter.

(2) The demineralized water makeup and gland seal systems (1000 gpm) can balance the loss of water for ruptures up to 4.3 inches in diameter.

(3) The fire-water supply alone can balance the loss of water for ruptures up to 6 inches in diameter.

In the cases above, the emergency pumps will not lose suction.

By reference to the previous discussions, certain ruptures below the elevation of the outlet lines can be handled by the emergency water-supply system even though suction to the emergency coolant pumps is lost.

(1) The fire-water supply alone can maintain adequate emergency coolant flow after the loss of suction to the emergency coolant pumps for ruptures up to 6.3 inches in diameter.

(2) The fire-water supply in conjunction with the demineralized water makeup and gland seal systems can maintain adequate emergency coolant flow after the loss of suction to the emergency coolant pumps for ruptures up to 8 inches in diameter.

In accordance with the foregoing, large ruptures in only the following nozzles, or penetrations on the reactor vessel below the outlet lines or in the corresponding inlet and outlet coolant lines cannot be adequately handled by the emergency water supply system.

Two 23-inch ID coolant inlet

Four 15-inch ID coolant outlet

Two 10-inch flush water connections.

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VII. INSTRUMENTATION AND CONTROL

- VII-1 Control Methods and Philosophy
- VII-2 Nuclear Instrumentation and Control
- VII-3 Reactor Control
- VII-4 Reactor and Plant Safety System
- VII-5 Process Instrumentation and Control
- VII-6 Radiation Monitoring System (RMS)
- VII-7 Data Processing System
- VII-8 ATR System Transient Behavior

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VII-1. CONTROL METHODS AND PHILOSOPHY

Control methods and philosophy adopted for the design of the ATR instrumentation are based on providing a high integrity system capable of controlling the facility in a safe manner for all required modes of operation. The several unique design features of the ATR to improve experiment performance also influence the instrumentation design. These features include:

- (1) Flux trapping to peak the thermal flux in the experiment area and also isolate the experiment from thermal flux variation (burn-up) in the fuel region.
- (2) Undisturbed axial flux distribution caused by control rod motion during a cycle.
- (3) Lobe power shift and control capability during a cycle.
- (4) Large fraction of core neutrons obtained from the reflector.

The ATR control system can be naturally divided into two parts: nuclear instrumentation and process instrumentation. Each part is independent except for reactivity feedback effects of coolant temperature and reactor kinetics.

The controlled parameters are monitored by multiple instrumentation channels or by backup devices which have a known relationship to each other. All corrective and safety actions can be initiated by any single parameter since coincident circuitry is not used in the ATR.

The control system also provides graded corrective actions to reduce the frequency of safety system action.

VII-2. NUCLEAR INSTRUMENTATION AND CONTROL

1. GENERAL

The four basic functions of the nuclear instrumentation and control system are:

- (1) To start up and operate the reactor at fixed conditions throughout the core cycle.
- (2) To adjust power distribution.
- (3) To provide safe and reliable operation from the source level to the operating level.
- (4) To shut the reactor down without damage in the event of accidents, malfunctions, or operation outside of specified conditions.

The ATR nuclear instrumentation system is based on Oak Ridge designed equipment. This instrumentation uses electron tubes and circuitry for compatibility with the present equipment, operating, and maintenance experience of the MTR and ETR. The degree of reliability of this equipment and circuitry has been established at MTR and ETR. The system consists of three groups of neutron detectors and associated channels suitable for operation in the source, start-up, intermediate, and power level ranges. At least one decade of overlap is provided between two groups of channels to improve system reliability. The detector ranges, including overlap, are shown in Figure VII-2-1.

Duplicate channels, including detectors circuitry and instruments, are provided as a minimum for each of the three groups to provide independent action. Two fission chambers are used in the source and start-up ranges. Two compensated ion chambers are used in the Log-N period range (intermediate), and four compensated ion chambers are used in the power level range. The nuclear instrumentation is calibrated by a total heat balance and measurements in each quadrant in the power level range and by foil activation in the start-up or intermediate range. A neutron source that provides a minimum readable count rate of one cps for the start-up channels with an unirradiated reflector is necessary to override a start-up interlock. An ample source of photoneutrons is available for start-up from gamma-beryllium reactions after the reactor has been operated at power.

In addition to the above neutron detector channels, a N-16 system has been provided to monitor lobe power changes and axial power shifts that can be caused by xenon perturbations.

A fission break monitor is also provided.

FIG. VII-2-1 POWER LEVEL INSTRUMENT RELATIONSHIPS.

2. NEUTRON DETECTION

2.1 Start-Up Instrumentation

The start-up instrumentation consists of two low-level, log-count-rate-meter circuit channels as shown in Figure VII-2-2. Each channel has a fission chamber sensitive to the low neutron densities in the clean cold reactor at source power, preamplifier, linear amplifier, log-count-rate meter, and recorder.

FIG. VII-2-2 NUCLEAR INSTRUMENTATION.

Logarithmic counting rate meters and recorders to indicate the rate of neutron production are energized by pulses from the fission chambers. At start-up, these chambers are initially located in a position near the core midplane. As the power level increases, the chambers are withdrawn by electric motors to other predetermined positions to keep the recorders and meters on scale and prevent damage.

The start-up channels range from source level (about $10^{-11}N_F$) to $10^{-5}N_F$ in adjustable increments of up to four decades. The fission chamber has a range of 1.5×10^0 to 1.5×10^5 n/cm²-sec. The neutron sensitivity of the fission chamber

is 0.7 counts/sec per n/cm^2 -sec. After amplifying the pulses from the chamber, the signal is changed from a linear to a logarithmic function and recorded. Since the useful working range of the count rate meter extends to only about 10^4 counts per second, provisions for increasing the system neutron flux range are made by an instrument drive mechanism which allows the operator to electrically withdraw the chamber to a predetermined position for further useful operation. These chambers operate at 300 dc volts from a power supply regulated to $\pm$ one percent.

2.2 Log-N Period Channels

The two Log-N period channels use compensated ion chambers to give reliable measurements of the neutron flux over a wide range in the presence of intense gamma radiation. The channel arrangements are shown in Figure VII-2-2.

The chamber output current ranges from 5×10^{-11} to 1.5×10^{-4} amperes and has a neutron sensitivity of 2.2×10^{-14} amperes per n/cm^2 -sec. This signal feeds a logarithmic amplifier. The output of this amplifier is proportional to the logarithm of the neutron flux. The period amplifiers are part of the safety system associated with the sigma bus. The period amplifiers are coupling devices between the Log-N amplifiers and the sigma amplifiers (scram input device). The period amplifiers produce a voltage proportional to the rate of change of the logarithm of the neutron flux, and convert the time rate of change signal to a voltage proportional to the period involved. This signal is transmitted to the sigma amplifiers. When this increases to a preset voltage representing a one-second period, an electronic scram results.

The signal from the Log-N amplifiers are fed to Log-N and period recorders.

2.3 Neutron Level Channels

The four neutron level channels use four compensated ion chambers. About four decades of power, 0.015 to 150 percent of rated power, can be measured. The output current from each ion chamber is transmitted to an amplifier. The linear amplifier output is transmitted to both a sigma amplifier and a power level recorder. The channel arrangements are shown in Figure VII-2-2.

Range switches provide either low power or high power signals to each linear amplifier. Interlocks control range switch.

An increase in flux increases the output voltage to the sigma bus until the trip voltage is reached and an electronic scram occurs. Loss of instrumentation power also causes a scram.

The amplifier signals are also fed to recorders.

2.4 Sigma Bus System

The sigma bus responds to the largest of the various safety signals. The instrument signals are fed to this bus by special sigma amplifiers in which the bus is connected to the cathode of a triode whose grid accepts the signal from the preamplifier. An increase in signal causes the preamplifier grid and the sigma amplifier triode grid to swing positive, making the sigma bus more positive. If this action has occurred in only one of the sigma amplifiers, the output tube cathodes of the other amplifiers are also carried positive, and the tubes tend to cut off. One of these amplifiers can take over from the first amplifier only if its grid is driven more positive than the first grid. Thus, the sigma bus is an auction

bus responding to the highest bidder of the various grid potentials. This predominance of auction effect over summation effect prevents electronic scrams initiated by the accumulation of small effects.

2.5 Deviation Galvanometer

A deviation galvanometer on the operating console indicates power error to the operator at various power levels. Three ranges and bias control allow operation over the power range of 1 to 100 percent of rated power. A selector switch is provided on the console for selection of the proper power range and sensitivity. The channel arrangement is shown in Figure VII-2-3. A signal from a compensated ion chamber is fed to a galvanometer through the selector switch. A bias voltage is used to limit the range of the circuit and produce an error signal.

FIG. VII-2-3 POWER LEVEL DEVIATION MONITOR.

2.6 Servo System

Two additional compensated ion chambers are used in the automatic servo system for power regulation. These chambers are in a voltage divider network and act in a similar manner to a resistance that varies with the magnitude of the flux field in which it is located.

2.7 Neutron Detector Locations

The location of the various neutron detectors is shown in Figure VII-2-4 together with the designated instrument thimble number.

2.8 Nuclear Instrumentation Nonlinearities

The background radiation, caused by gamma and photoneutrons, is high in the vicinity of the neutron detectors. It was, therefore, necessary to evaluate

FIG. VII-2-4 NUCLEAR INSTRUMENTATION THIMBLE POSITIONS.

the effect of nonlinearities, caused by background radiation, in the nuclear instruments over the range from start-up to operating power.

The fission chambers will be positioned close to the core during clean start-up. The reflector shim control drum position will affect the thermal flux at the detector position when the fission chambers are inserted, and some shadowing of the chambers will result. This has been experienced in the operation of the ATR critical experiment when the outer shim controls have been moved in the vicinity of a fission chamber. However, at higher power levels, with the chambers withdrawn about a foot above the reflector, the thermal flux at the detector is caused almost entirely from the slowing down of fast neutrons that leak from the core.

Since this fast flux leakage is relatively insensitive to control element effects, the detector characteristics at a foot or more above the reflector are free of shadowing. Accordingly, reactor start-up procedures with the clean core have been developed to eliminate false count-rate meter indications.

The start-up instrumentation will receive a nonlinear signal at low levels in the presence of high background radiation. These instruments are sensitive to gamma radiation, and to the photo neutrons produced in the beryllium by gamma, neutron reactions. Since neither of these background sources vary directly as power level, nonlinearities will occur in the detector reading. An evaluation of the start-up instrumentation in high radiation backgrounds has been made. Figure VII-2-5 shows the effect of this background on the inverse counting rate. The deviations from linearity caused by the above described background radiation are not considered a problem in starting up a fully fueled reactor.

Reactor operation above SAG (0.1 percent full power) with the presence of high background radiation can introduce significant power level instrumentation nonlinearities. As shown in Table VII-2-1, the effects of these nonlinearities are greatly reduced by calibration at or near the desired operating level.

FIG. VII-2-5 APPROACH TO CRITICALITY.

TABLE VII-2-1

EFFECT OF CALIBRATION ON INSTRUMENT READING

Power	Detector Reading	
	Calibration at N_L	Calibration at N_F
$1.3 N_F$	$1.2409 N_F$	$1.29905 N_F$
N_F	$0.95465 N_F$	N_F
N_L	N_L	$1.31388 N_L$
SAG ($0.1N_L$)	1.4123 SAG	4.16732 SAG

3. N-16 MONITORING SYSTEM

The N-16 system monitoring system, arranged as shown in Figure VII-2-6, is an array of 10 water sample tubes located in appropriate positions in the

FIG. VII-2-6 MONITORING INSTRUMENTATION.

reactor core to bring samples from lobe regions to beta detectors. This system is independent of primary coolant water since demineralized makeup water is used on a once through basis. The water flows up through an inner tube, reaches the core area where fast neutrons, proportional to lobe power, react with O-16 (n, p) to form N-16. The N-16 decays (beta-gamma) with a half-life of 7.4 seconds. The N-16 activated water takes about 15 seconds to travel from the core to the detectors. In addition to providing interlobe power comparisons, two of the sample tubes are arranged in the upper and lower sections of the core, respectively, to provide an indication of core power fluctuations in the axial direction that could occur if xenon perturbations should arise. Reliable flow control is provided to prevent the detectors from drifting off calibration.

The ten detector signals are fed to linear picoammeters that feed signal converters providing voltage signals to potentiometric-type recorders. The recorders supply signals to the data processing system which calculates lobe power allowing the reactor operator to maintain predetermined lobe power distribution.

The detectors are calibrated by foil activation and thermodynamic measurement techniques.

4. FISSION BREAK MONITORS

Two fission break monitoring channels are provided to detect a release of fission products into the primary coolant system. Each channel consists of a gamma ionization chamber, a picoammeter, and a recorder. The detectors are located at the primary system outlet piping header. Delay time is several seconds.

VII-3. REACTOR CONTROL

1. CONTROL PHILOSOPHY

The ATR reactor control is based on the capability of producing and maintaining a power shift among the various lobes and maintaining the reactor power level at $\pm 1/2$ percent of full power throughout the cycle. The control system is designed to provide interlocks and graded corrective action to minimize safety system action.

2. METHODS OF CONTROL

The ATR control system enables the operator to maintain the proper power level and power distribution among the core lobes. It can automatically compensate for reactivity changes due to fuel burnup, fission product formation, depletion of burnable poison, and other sources of reactivity perturbation. Automatic control maintains reactor power at $\pm 1/2$ percent of full power. This control is achieved through poison control rods in the neck and reflector regions of the reactor core.

2.1 Outer Shim Control Cylinders

The outer shim control cylinders are used for reactivity shimming and flux shifting. These cylinders, operating in pairs, are manually selected and controlled by the reactor operator. These control elements may be operated as a group containing any number of pairs from one to eight whenever the withdrawal permissives are energized. All elements fully insert automatically after a time delay following a scram. A reverse will insert all elements until the signal clears.

2.2 Neck Shim Control Rods

The neck shim rods provide for reactivity shimming and, to a lesser degree, flux shifting. Neck shim rods can be withdrawn only under manual control either individually or as a group of any number below $10^{-5}N_F$. When the reactor is above $10^{-5}N_F$, withdrawal is limited to one rod at a time. The neck shim control rods will fully insert automatically after a time delay following a scram. A reverse will automatically insert the neck shims until the signal clears.

2.3 Servo or Automatic Control

The two regulating neck rods are driven by servo motors under control of the servo channels. These rods are driven at speeds up to 50 inches/sec. Only one of the two regulating rods and associated channel is used at any one time (the other is fully withdrawn) to limit the ΔK available to the automatic control system to approximately 0.4 percent maximum. A selector switch is available to the operator to choose the desired regulating rod and channel that is to be placed in service. The regulating rod can be placed in manual or automatic (servo) control as desired, provided the reactor run interlock is engaged.

Manual shimming is used to keep the active regulating rod in its normal operating range to compensate for relatively slower reactivity changes from fuel burnup, xenon, etc. Figure VII-3-1 indicates the automatic shimming and restriction interlocks as a function of active regulating rod position. The normal regulating rod range is from 20 to 42 inches withdrawn (range 5). If the regulating rod is withdrawn beyond 42 inches (range 6), it can be restored to range 5 by manually withdrawing shims. Shim rod withdrawal is prevented (interlock) if the regulating rod is in the range of 0 to 20 inches withdrawn (ranges 1 to 4) and the reactor is in Run Auxiliary (Servo control). Automatic outer shim insertion occurs in ranges 1 to 3 until the regulating rod position is restored to range 5. All the outer shims insert in ranges 1 and 2 (automatic cylinder insert). In range 3, a selected group of outer shims is automatically inserted (automatic preferred cylinder insert).

FIG. VII-3-1 REGULATOR ROD CONTROL RANGES.

The ATR automatic (servo) control system arrangement, shown in Figure VII-3-2, is similar to that used in the MTR. The servo system ion chamber and the motor operated rheostat (MOR) are connected in series across a positive and negative power supply, so that an error signal appears at the electrical junctions. A change in either the flux (chamber signal) or the MOR produces an error voltage signal. The resulting signal feeds the servo preamplifiers and regulating rod control system for a correction in the reactor power level to the desired level. The power range setting for the MOR setpoint is adjustable from 2.5 to 250 MW(t).

A test program was performed on the automatic control system electronic components to (a) develop the lead-lag network in the controller for optimum regulation relative to the reactor kinetic characteristics and the regulating rod drive dynamic response, (b) test the electronic components to assure proper

FIG. VII-3-2 POWER LEVEL SERVO SYSTEM.

integral performance as a system, and (c) determine the limit of adjustment available to the reactor operator. The results of this program showed the control instrumentation performed satisfactorily within the closed loop system.

2.4 Automatic Power Reduction

Certain abnormal conditions occurring in the reactor, the experiments, or the process system initiate power reductions. Recovery from such an action is not permitted until the fault is cleared and safe operation assured. These reduction methods are described from the least to most severe action.

(1) Setback -- A control system action to reduce the reactor power level by automatic lowering of the setpoint of the automatic control system. The MOR setpoint is continuously driven downward toward N_L as long as the initiating signal persists. This control action consists of a fast and slow setback which requires 90 seconds and 8 minutes, respectively, to lower the MOR from N_F to N_L . When the trouble signal clears, the servo will respond to the operator's demand for return to the desired servo setpoint. In the event that the reactor is not in automatic (Run auxiliary) control, the setback signal will initiate a reverse.

(2) Reverse -- A control system action which inserts all shim rods and active regulating rod. The regulating rod is inserted at a speed of five inches per second. The insertion continues until the initiating signal clears. The regulating rod inserts until reaching range 1.

(3) Fast Recovery Scram -- Fast recovery scram reduces reactor power quickly by scrambling the selected safety rod (withdrawal speed of nine inches per minute). After a fast recovery scram signal clears, the safety rod can be withdrawn. The control system interlocks prevent any shim rod withdrawal until the fast recovery scram safety rod reaches its upper limit switch. The fast recovery scram is used only for signals from experiments.

(4) Scram -- This action releasing all safety rods is described in the section on Reactor Safety Systems.

3. CONTROL INTERLOCKS

The ATR control system contains a system of interlocks designed to require the operator to start up and operate the reactor in a manner to ensure safe conditions prevail at all times. These interlocks include operational sequence interlocks, console operation interlocks, safety rod interlocks, outer shim cylinder interlocks, neck shim interlocks, regulating rod interlocks, and control panel instrument interlocks. A procedure panel is also provided with visual indication (lights) to aid the operator during start-up.

3.1 Operational Sequence Interlocks

The following interlocks are provided to ensure proper sequence of operations:

(1) The console key switch must be on before any rod withdrawal is permitted. A horn and "Reactor On" lights are activated by this switch.

(2) A start-up channel recorder must be indicating on scale before safety rods can be withdrawn.

(3) Safety rod upper limit switches must be engaged before shim rods can be withdrawn.

(4) The control rods operate with a time controlled intermittent on-off type operation whenever the reactor power level is less than $10^{-5}N_F$. The timer allows rod withdrawal for only one second out of each five seconds. Full withdrawal of the safety rods requires about 45 minutes. Above $10^{-5}N_F$, the above timer interlock and the start-up channel requirement interlock are bypassed. The neck shims can only be withdrawn individually above $10^{-5}N_F$.

(5) When reactor power reaches SAG, the two fission chambers are automatically withdrawn if the operator has failed to withdraw the chambers.

(6) The above SAG interlock also starts a timer that will automatically shift the lower level electronic scram trip from $1.2 N_L$ to the operating power level trip point provided the power has not reached $1.2 N_L$ in 11.5 seconds. A scram will occur if $1.2 N_L$ is reached in less than 11.5 seconds. This interlock is essentially a period backup safety to prevent an excessively fast start-up from low power levels.

(7) Above SAG, the operator can push the Run switch which results in the following action:

(a) The 30-second period control rod withdraw prohibit is bypassed.

(b) Servo control permit is available.

(c) The run switch relay is locked in.

(d) Procedure panel indication is given.

Failure to push the run button above SAG prevents servo operation and limits manually controlled power increase to a period greater than 30 seconds.

(8) The reactor will scram above $3N_L$ if not in the Run regime.

(9) The run relay will drop out causing the reactor to return to the Start regime when the reactor power level drops below SAG. In addition, the MOR setpoint is automatically reset to N_L .

3.2 Console Operation Interlocks

The following interlocks are provided to prevent operator error:

(1) An interlock prevents simultaneous withdrawal and insertion signals with the insertion signal overriding the withdrawal signal.

(2) Setpoint interlocks include:

A 30-second period prevents any further withdrawal of the rods when the reactor is in the start regime.

A 7-second period prevents any further control rod withdrawal in the run regime.

A 5-second period causes a reverse.

A 1-second period causes an electronic scram.

3.3 Safety Rod Interlocks

The ATR safety rod system is interlocked to prevent operation without adequate shutdown margin. The following interlocks apply to the safety rods:

(1) If two safety rods insert at any time, the reactor will scram.

(2) Above $3N_L$, a fast recovery scram will be followed by a scram if the signal has not cleared in 60 seconds.

(3) Below $3N_L$, a fast recovery scram will cause a relay scram.

3.4 Outer Shim Cylinder Interlocks

The following interlocks affect the outer shim cylinders:

(1) If the regulating rod is in range 1 or 2, all outer shim cylinders insert until the regulating rod returns to range 5. Shim withdrawal is not permitted.

(2) If the regulating rod is in range 3, the selected group of outer shim cylinders insert until the regulating rod returns to range 5. Shim withdrawal is not permitted.

(3) If the regulating rod is in range 4, shim withdrawal is not permitted.

(4) Following a relay or electronic scram, the outer shim cylinders remain stationary for a timed period. If the safety rods reach their upper limit prior to the timed period, no shim insertion occurs. If the safety rods are not at their upper limit switches at the end of the timed period, all outer shim cylinders are fully inserted. A manual or instrument power failure scram inserts the shims immediately. Shim withdrawal is not permitted until safety rods are at the upper limit.

(5) Automatic transfer to battery-backed ac power supply to drive motors occurs in the event of total failure of diesel-commercial power.

3.5 Neck Shim Rod Interlocks

The neck shim rods are interlocked as follows:

- (1) Only one rod can be withdrawn at a time when the reactor power level is above $10^{-5}N_F$.
- (2) Shim insertion following a scram is the same as mentioned above for the outer shim cylinders.
- (3) Automatic transfer to battery-backed ac power supply to drive motors occurs in the event of total failure of diesel-commercial power supply.

3.6 Regulating Rod Interlocks

Regulating rod interlocks have been covered in the preceding sections.

3.7 Control Panel Instrument Interlocks

A system of test blocks is provided which permits the operator to disconnect certain instruments and circuits for test and repair. Each pressure switch, recorder contact, or other contact-making device which can initiate a reactor power reduction is connected to the power-reduction relays through a test block. All test blocks are mounted on the Reactor Control Panel, and are visible from the control console. Certain combinations of test block plugs must be in place to permit a given operation. Table VII-3-1 is a list of the test blocks with their functions, locations, and backups.

Each power-reduction relay initiating contact closes whenever its corresponding parameter (flow, temperature, etc) exceeds a predetermined setpoint. This contact closure actuates relays for the proper type of power reduction (scram, reverse, etc) provided the corresponding test plug is in place. The removal of a test plug disconnects the parameter from its power-reduction function. Adjustment or repair of the initiating circuit can then be made. However, since each parameter is either double tracked or backed up with a related parameter, protection is maintained. If the backup or double track test plug is mistakenly removed, at the same time, a relay scram will automatically occur.

Whenever a test plug is removed, the annunciator for the parameter associated with the test block is energized. The test plug removal is also recorded by the data processing system.

In a few special cases, the test blocks function differently from the manner indicated above. Each safety rod upper limit switch is provided with a test block which short-circuits the normal path of the limit switch (when pulled) and disconnects one side of the limit switch for test purposes. The ATR has five safety rods installed and provisions for the installation of three additional rods if ever required. Accordingly, the Reactor Control Panel has eight safety rod upper limit switch test blocks. The test plugs must be removed from the three vacant positions to allow shim rod withdrawal permit. When two additional test plugs are removed, a scram will occur. This allows operation with only one test plug removed for one of the five installed rods. The position of this rod can be determined from the continuous position readout.

TABLE VII-3-1

TEST BLOCKS

Test Block Number	Parameter	Double Track or Backup Test Block Number
TB-1	Log N Channel 4	TB-2
TB-2	Log N Channel 5	TB-1
TB-3	Neutron Level Channel 7	TB 4, 5, and 6
TB-4	Neutron Level Channel 9	TB 3, 5, and 6
TB-5	Neutron Level Channel 8	TB 3, 4, and 6
TB-6	Neutron Level Channel 6	TB 3, 4, and 5
TB-7	N-16 Channel NE Inner Lobe	[a]
TB-8	N-16 Channel SE Inner Lobe	[a]
TB-9	N-16 Channel SW Inner Lobe	[a]
TB-10	N-16 Channel NW Inner Lobe	[a]
TB-11	N-16 Channel N Outer Lobe	[a]
TB-12	N-16 Channel E Outer Lobe	[a]
TB-13	N-16 Channel S Outer Lobe	[a]
TB-14	N-16 Channel W Outer Lobe	[a]
TB-17	Reactor Period Channel 4	TB-18
TB-18	Reactor Period Channel 5	TB-17
TB-19	Start-up Channel 1	TB-20
TB-20	Start-up Channel 2	TB-19
TB-21	Primary Coolant Flow Quadrant 1	TB-22
TB-22	Primary Coolant Flow Quadrant 1 Backup	TB-21
TB-23	Primary Coolant Flow Quadrant 2	TB-24
TB-24	Primary Coolant Flow Quadrant 2 Backup	TB-23
TB-25	Primary Coolant Flow Quadrant 3	TB-26
TB-26	Primary Coolant Flow Quadrant 3 Backup	TB-25
TB-27	Primary Coolant Flow Quadrant 4 Backup	TB-28
TB-28	Primary Coolant Flow Quadrant 4	TB-27
TB-29	Primary Coolant Temperature Rise Quadrant 1	TB-3
TB-30	Primary Coolant Temperature Rise Quadrant 2	TB-4
TB-31	Primary Coolant Temperature Rise Quadrant 3	TB-5

TABLE VII-3-1 (cont.)

TEST BLOCKS

Test Block Number	Parameter	Double Track or Backup Test Block Number
TB-32	Primary Coolant Temperature Rise Quadrant 4	TB-6
TB-33	Water Power Quadrant 1	None
TB-34	Water Power Quadrant 2	None
TB-35	Water Power Quadrant 3	None
TB-36	Water Power Quadrant 4	None
TB-37	Primary Surge Tank Level	TB-38
TB-38	Primary Surge Tank Level -- Backup	TB-37
TB-39	Reactor Inlet Water Temperature	TB-40 thru TB-43
TB-40	Primary Coolant Outlet Temperature Quadrant 1	TB-39
TB-41	Primary Coolant Outlet Temperature Quadrant 2	TB-39
TB-42	Primary Coolant Outlet Temperature Quadrant 3	TB-39
TB-43	Primary Coolant Outlet Temperature Quadrant 4	TB-39
TB-44	Reactor Inlet Primary Pressure	TB-45
TB-45	Reactor Inlet Primary Pressure Backup	TB-44
TB-47	Reactor Instrument Air Pressure	TB-48
TB-48	Reactor Instrument Air Pressure Backup	TB-47
TB-50	Low Voltage Commercial System Bus A Phase A-B	TB-51
TB-51	Low Voltage Commercial System Bus A Phase B-C	TB-50
TB-52	Low Voltage Commercial System Bus B Phase A-B	TB-53
TB-53	Low Voltage Commercial System Bus B Phase B-C	TB-52
TB-54	Low Voltage Diesel Phase A-B	TB-55
TB-55	Low Voltage Diesel Phase B-C	TB-54
TB-56	Earthquake	TB-57
TB-57	Earthquake Backup	TB-56
TB-58	Emergency Coolant Pumps Flow	TB-83, TB-84

TABLE VII-3-1 (cont.)

TEST BLOCKS

Test Block Number	Parameter	Double Track or Backup Test Block Number
TB-63	AC Emergency Coolant Pump M10	TB-64
TB-64	AC Emergency Coolant Pump M10 Recirculation -- Backup	TB-63
TB-65	DC Emergency Coolant Pump M11 Recirculation	TB-74
TB-66	Upper Limit Switch Safety Rod SRNE	[b]
TB-67	Upper Limit Switch Safety Rod SRE	[b]
TB-68	Upper Limit Switch Safety Rod SRSE	[b]
TB-69	Upper Limit Switch Safety Rod SRS	[b]
TB-70	Upper Limit Switch Safety Rod SRSW	[b]
TB-71	Upper Limit Switch Safety Rod SRW	[b]
TB-72	Upper Limit Switch Safety Rod SRNW	[b]
TB-73	Upper Limit Switch Safety Rod SRN	[b]
TB-74	DC Emergency Coolant Pump M11 Recirculation -- Backup	TB-65
TB-79	Fission Break Detector -- Channel 13	TB-80
TB-80	Fission Break Detector -- Channel 14	TB-79
TB-81	Reactor Vessel Liquid Level	TB-82
TB-82	Reactor Vessel Liquid Level -- Backup	TB-81
TB-83	AC Emergency Coolant Pump M10	TB-58[c]
TB-84	Recirculation	
TB-84	DC Emergency Coolant Pump M11 Recirculation	TB-58[c]
SD-TB-1	Shutdown Test Block-Safety Rods	SD-TB-2, SD-TB-3
SD-TB-2	Shutdown Test Block-Outer Shims	SD-TB-1, SD-TB-3
SD-TB-3	Shutdown Test Block-Neck Shims	SD-TB-1, SD-TB-2

[a] A scram will occur when less than three of the N-16 test plugs (TB-7 through TB-14) are in place.

[b] A scram will occur when two upper limit switch test plugs are removed, provided the three test blocks have been pulled for the rod positions which do not have safety rods installed.

[c] Effective only during flux-mapping operation.

No test plug is provided for the fast recovery scram rod upper limit switch.

Shutdown test blocks are installed to permit testing of the safety rod mechanisms during reactor shutdown. Pulling one of these test blocks permits a single safety rod to be raised or lowered under manual control. Pulling of two of these test blocks will initiate a scram.

One of each safety rod lower limit switch or seat switch must be operable for rod withdrawal permit.

The integrity of the looped safety system buses may be tested by pressing the loop bus test button while the reactor is in operation, and observing the three adjacent indicating loops. If one of the lamps goes out when the button is pressed, the corresponding bus is broken and should be repaired. The above test is not valid if any test plug is pulled or any power reduction occurs during the test.

4. LOW-POWER, OPEN-HEAD OPERATION

The ATR is designed to operate at low power (2.5 MW or less) with the primary system at atmospheric pressure and with the refueling ports removed. The purpose of this operation is to allow flux mapping of the core.

The reactor power can be automatically controlled when operation is above SAG, but manual control must be used at the lower power levels. The reactor is cooled during this type of operation by utilizing the flow from both emergency coolant pumps and the utility cooling-water system to remove heat from the secondary side of the primary heat exchangers. Special protection is provided during this type of operation.

VII-4. REACTOR AND PLANT SAFETY SYSTEM

1. GENERAL

The criterion for design of the ATR safety system is that an abnormal condition with a reasonable probability of occurring will not result in fuel element melting at the hottest spot in the core. For any condition that could lead to fuel melting, there are at least two completely independent instrumentation detection, safety system circuits to cause a scram. Each circuit is capable of uninterrupted operation during reactor operation. Failure of any component in either circuit does not impair the proper operation of the other circuit. This backup scram philosophy is the same as that of the MTR and ETR.

There are five safety rod magnets under the control of the safety signals. The magnet current is drawn from a separate rectifier, fed from a power line separate from the magnet amplifier. This permits interruption of magnet power without affecting the amplifier power and makes possible two kinds of scram:

- (1) Electronic scram by amplifier action
- (2) Relay scram by interruption of magnet power.

Each magnet is controlled by two amplifiers connected to share the magnet current. A short to ground of the sigma bus will cut off magnet current and cause a scram. Either amplifier can assume total current load if failure occurs in the other unit.

2. TYPES OF ACCIDENTS

The type of accidents studied have been classified as reactor equipment malfunctions, plant equipment malfunctions, and experiment failures.

2.1 Malfunction of Reactor Equipment

Power level instrumentation contains four independent channels to preclude failure of one circuit from preventing information transmittal to the safety system. The safety system also has period information available to prevent core damage. The period instrumentation is backed up by a low-level over-power trip and timer circuitry.

2.2 Plant Equipment Malfunction

The plant is protected against loss of power by scram signals initiated by a loss of commercial power, loss of diesel power, and by a low-flow backup signal from the quadrant primary coolant flow meters.

The reactor is protected from loss of pressure by primary coolant system pressure measurements backed up by pressurizer liquid level measurement.

The reactor is protected from excessive coolant temperature by inlet coolant temperature measurement backed up by the reactor outlet coolant temperature.

The reactor is protected from excessive or low flow by primary coolant flow measurements backed up by additional flow measurements.

Low reactor coolant level conditions are protected against by vessel level measurements and are double tracked.

Instruments and detectors are arranged to ensure that single component failures will not prevent a reactor scram if required. Any single failure that could prevent the safety system from responding to a scram, initiates a scram signal itself. An example of this is loss of power to instruments.

Manual scram buttons are provided for the operator. Certain conditions outside the plant may require that the operator secure the plant. Automatic scram actions are also provided to prevent uncontrolled operation of the plant. These are:

- Seismic activity

- Loss of instrument air

- Loss of diesel power

- Loss of safety system voltage

- Loss of instrument system voltage

- Simultaneous removal of both test blocks of a double tracked or backed up pair of parameters.

AC battery-backed power is available to the shim rod insert buses when the normal power source from the diesel-commercial bus is lost. A push button is located on the reactor console to cycle the bus switching circuitry for testing purposes.

2.3 Experiment Failures

Experiments to be inserted into the ATR must be designed and operated to prevent experiment failure and to ensure that failures and abnormal conditions will not cause reactor transients that exceed the capability of the safety system to protect the reactor. The experiment designs must accommodate the normally expected transients of power, pressure, flow, and temperature. Protection of the experiment is accomplished through a reactor power reduction. The degree of protection afforded the experiments includes a relay scram, fast recovery scram, reverse, and setback. The experiments are designed within the capabilities of the above power reduction rates.

3. SAFETY SYSTEM ACTION

The two types of scrams provided at the ATR both interrupt power to the electromagnetic clutch holding the five safety rods in the cocked (withdrawn) position. The electronic scram is the faster of the two types, as the delay in the electronic circuitry is less than 1 msec. The input signal is initiated by a change in neutron flux that alters the bias on the magnet amplifiers and releases all safety rods. The safety rods begin to insert in less than 25 msec. The outer shim cylinders and neck shims are also automatically inserted to the full-in position after a time delay. A relay scram releases all safety rods when energized parameter contacts de-energize the relay in the power circuit to the magnet amplifiers. Total release time is about 60 msec.

4. SAFETY SYSTEM INPUTS

All input signals to the reactor safety system are scram signals coming from the reactor nuclear instrumentation, the process system instrumentation, and experiments. Tables VII-4-1 and -4-2 present the detailed list of functions causing scram action from the reactor system and process system, respectively. Table VII-4-3 lists the special functions used to scram the reactor during low-power, open-head operation. Table VII-4-4 lists the corrective action and safety functions available to experiments. Table VII-4-5 lists miscellaneous functions and their respective corrective and safety actions. The tables also include the backup device where applicable. Figure VII-4-1 is a schematic of the safety system arrangement.

TABLE VII-4-1

NUCLEAR CONTROL AND SAFETY ACTIONS

Item Number	Nuclear Quantities	Device Identification	Setback	Reverse	Electronic Scram	Relay Scram	Comments
1	High Neutron Flux Level	Level Channel #7	1.1 N _F		1.2 N _F	1.2 N _F	
2	High Neutron Flux Level	Level Channel #9	1.1 N _F		1.2 N _F	1.2 N _F	
3	High Neutron Flux Level	Level Channel #8	1.1 N _F		1.2 N _F	1.2 N _F	Pulling of any 3 test plugs causes a scram
4	High Neutron Flux Level	Level Channel #6	1.1 N _F		1.2 N _F	1.2 N _F	
5	Reactor Period	Log N Channel #4		5 sec	1 sec		
6	Reactor Period	Log N Channel #5		5 sec	1 sec		
7	Neutron Flux Level	Level Channel #7	1.1 N _L		1.2 N _L	1.2 N _L	Cut out 11.5 seconds after exceeding "Sag"
8	Neutron Flux Level	Level Channel #9	1.1 N _L		1.2 N _L	1.2 N _L	Cut out 11.5 seconds after exceeding "Sag"
9	Neutron Flux Level	Level Channel #8	1.1 N _L		1.2 N _L	1.2 N _L	Cut out 11.5 seconds after exceeding "Sag"
10	Neutron Flux Level	Level Channel #6	1.1 N _L		1.2 N _L	1.2 N _L	Cut out 11.5 seconds after exceeding "Sag"
10a	Fuel Element Rupture	Fission Break Detector #1				Yes	
10b	Fuel Element Rupture	Fission Break Detector #2				Yes	
11	N-16 Power Level NE Inner Lobe	N-16 Channel #1	1.1 N _F			1.2 N _F	Selectable to Setback or Scram from 3 (selected out of 8)
12	SE Inner Lobe	N-16 Channel #2	1.1 N _F			1.2 N _F	Selectable to Setback or Scram from 3 (selected out of 8)
13	SW Inner Lobe	N-16 Channel #3	1.1 N _F			1.2 N _F	Selectable to Setback or Scram from 3 (selected out of 8)
14	NW Inner Lobe	N-16 Channel #4	1.1 N _F			1.2 N _F	Selectable to Setback or Scram from 3 (selected out of 8)
15	N Outer Lobe	N-16 Channel #5	1.1 N _F			1.2 N _F	Selectable to Setback or Scram from 3 (selected out of 10)
16	E Outer Lobe	N-16 Channel #6	1.1 N _F			1.2 N _F	Selectable to Setback or Scram from 3 (selected out of 10)
17	S Outer Lobe	N-16 Channel #7	1.1 N _F			1.2 N _F	Selectable to Setback or Scram from 3 (selected out of 10)
18	W Outer Lobe	N-16 Channel #8	1.1 N _F			1.2 N _F	Selectable to Setback or Scram from 3 (selected out of 10)
19	High Neutron Flux Level	Log N Channel #4		3 N _L			Reactor in start regime or emergency fire-water shutoff valve GT-T-1-84 not fully open
20	High Neutron Flux Level	Log N Channel #5		3 N _L			Reactor in start regime or emergency fire-water shutoff valve GT-T-1-84 not fully open

TABLE VII-4-2

PROCESS CONTROL AND SAFETY ACTIONS

Item Number	Process Quantities	Device Identification	Setback	Relay Scram	Backup Device Number	Backup Item Number
21	Primary Coolant Quadrant #1 Flow -- High	FRS-1-1C and E	1.1 N	1.2 N	FS-1-11C and E	29
22 ^[a]	Primary Coolant Quadrant #1 Flow -- Low	FRS-1-1D and F	0.95 N	0.9 N	FS-1-11D and F	30
23	Primary Coolant Quadrant #2 Flow -- High	FRS-1-2C and E	1.1 N	1.2 N	FS-1-12C and E	31
24 ^[a]	Primary Coolant Quadrant #2 Flow -- Low	FRS-1-2D and F	0.95 N	0.9 N	FS-1-12D and F	32
25	Primary Coolant Quadrant #3 Flow -- High	FRS-1-3C and E	1.1 N	1.2 N	FS-1-13C and E	33
26 ^[a]	Primary Coolant Quadrant #3 Flow -- Low	FRS-1-3D and F	0.95 N	0.9 N	FS-1-13D and F	34
27	Primary Coolant Quadrant #4 Flow -- High	FRS-1-4C and E	1.1 N	1.2 N	FS-1-14C and E	35
28 ^[a]	Primary Coolant Quadrant #4 Flow -- Low	FRS-1-4D and F	0.95 N	0.9 N	FS-1-14D and F	36
29	Primary Coolant Quadrant #1 Flow -- High Backup	FRS-1-11C and E	1.1 N	1.2 N	FRS-1-1C and E	21
30 ^[a]	Primary Coolant Quadrant #1 Flow -- Low Backup	FRS-1-11D and F	0.95 N	0.9 N	FRS-1-1D and F	22
31	Primary Coolant Quadrant #2 Flow -- High Backup	FRS-1-12C and E	1.1 N	1.2 N	FRS-1-2C and E	23
32 ^[a]	Primary Coolant Quadrant #2 Flow -- Low Backup	FRS-1-12D and F	0.95 N	0.9 N	FRS-1-2D and F	24
33	Primary Coolant Quadrant #3 Flow -- High Backup	FRS-1-13C and E	1.1 N	1.2 N	FRS-1-3C and E	25
34 ^[a]	Primary Coolant Quadrant #3 Flow -- Low Backup	FRS-1-13D and F	0.95 N	0.9 N	FRS-1-3D and F	26
35	Primary Coolant Quadrant #4 Flow -- High Backup	FRS-1-14C and E	1.1 N	1.2 N	FRS-1-4C and E	27
36 ^[a]	Primary Coolant Quadrant #4 Flow -- Low Backup	FRS-1-14D and F	0.95 N	0.9 N	FRS-1-4D and F	28
37a	Quadrant #1 Differential Pressure -- High	DPS-1-21C		115 psig	DPS-1-22C, -23C, and -24C	37a, c, and d
37b	Quadrant #2 Differential Pressure -- High	DPS-1-22C		115 psig	DPS-1-21C, -23C, and -24C	37a, c, and d
37c	Quadrant #3 Differential	DPS-1-23C		115 psig	DPS-1-21C, -22C, and -24C	37a, b, and d
37d	Quadrant #4 Differential Pressure -- High	DPS-1-24C		115 psig	DPS-1-21C, -22C, and 23C	37a, b, and c
37e	None					
37f	None					
37g	None					
37h	None					
38	Quadrant #1 Temperature Rise -- High	TRS-1-11B and C	1.05 N	1.2 N	TRS-1-11B and C	52
39	Quadrant #2 Temperature Rise -- High	TRS-1-12B and C	1.05 N	1.2 N	TRS-1-12B and C	53
40	Quadrant #3 Temperature Rise -- High	TRS-1-13B and C	1.05 N	1.2 N	TRS-1-13B and C	54
41	Quadrant #4 Temperature Rise -- High	TRS-1-14B and C	1.05 N	1.2 N	TRS-1-14B and C	55

TABLE VII-4-2 (cont.)

PROCESS CONTROL AND SAFETY ACTIONS

Item Number	Process Quantities	Device Identification	Setback	Relay Scram	Backup Device Number	Backup Item Number
42	Quadrant #1 Water Power -- High	WRS-1-1B and C	1.1 N _P	1.2 N _P	---	
43	Quadrant #2 Water Power -- High	WRS-1-2B and C	1.1 N _P	1.2 N _P	---	
44	Quadrant #3 Water Power -- High	WRS-1-3B and C	1.1 N _P	1.2 N _P	---	
45	Quadrant #4 Water Power -- High	WRS-1-4B and C	1.1 N _P	1.2 N _P	---	
46[b]	AC Primary Emergency Coolant Pump MIO Recirculation -- Low	FS-1-20A or		210 gpm	FIS-1-23	46a
46a[b]	AC Primary Emergency Coolant Pump MIO -- Recirculation -- Low Backup	FIS-1-23 and		210 gpm	FS-1-20A	46
46b[a]	DC Primary Emergency Coolant Pump M11 Recirculation -- Low	FS-1-21A or		210 gpm	FIS-1-24	46c
46c[a]	DC Primary Emergency Coolant Pump M11 -- Recirculation -- Low Backup	FIS-1-24		210 gpm	FS-1-21A	46b
47[a]	Primary Surge Tank Level -- Low	LS-1-1C and D	N - 2 inches	N - 2-1/2 inches	LS-1-11C and D	48
48[a]	Primary Surge Tank Level -- Low Backup	LS-1-11C and D	N - 2 inches	N - 2-1/2 inches	LS-1-1C and D	47
49	Primary Surge Tank Level -- High	LS-1-1E and F	N + 2 inches	N + 2-1/2 inches	LS-1-11E and F	50
50	Primary Surge Tank Level -- High Backup	LS-1-11E and F	N + 2 inches	N + 2-1/2 inches	LS-1-1E and F	49
51	Reactor Inlet Water Temperature -- High	TRS-1-18B and C	128°F	145°F	(TRS-1-11B and C) (TRS-1-12B and C) (TRS-1-13B and C) (TRS-1-14B and C)	52 53 54 55
52	Primary Coolant Quadrant #1 Outlet Temperature High	TRS-1-11B and C	N + 3°F	N + 20°F	TRS-1-18B and C, DTRS-1-1B and C	51, 38
53	Primary Coolant Quadrant #2 Outlet Temperature High	TRS-1-12B and C	N + 3°F	N + 20°F	TRS-1-18B and C, DTRS-1-2B and C	51, 39
54	Primary Coolant Quadrant #3 Outlet Temperature High	TRS-1-13B and C	N + 3°F	N + 20°F	TRS-1-18B and C, DTRS-1-3B and C	51, 40
55	Primary Coolant Quadrant #4 Outlet Temperature High	TRS-1-14B and C	N + 3°F	N + 20°F	TRS-1-18B and C, DTRS-1-4B and C	51, 41
56	Primary Pressure at Reactor Inlet-High	PRS-1-31E and G	370 psig	375 psig	PRS-1-11E and G	58
57[a]	Primary Pressure at Reactor Inlet-Low	PRS-1-31F and H	340 psig	335 psig	PRS-1-11F and H	59
58	Primary Pressure at Reactor Inlet-High Backup	PRS-1-11E and G	370 psig	375 psig	PRS-1-31E and G	56
59[a]	Primary Pressure at Reactor Inlet-Low Backup	PRS-1-11F and H	340 psig	335 psig	PRS-1-31F and H	57
60	None					

[a] Automatically cut out at power less than 3N_L, reactor inlet pressure as measured by PRS-1-11J less than 30 psig, and emergency fire-water shutoff valve closed.

[b] Following adjustable time delay (0 to 10 seconds), to allow for automatic starting of second pump.

TABLE VII-4-3

CONTROL AND SAFETY ACTIONS FOR LOW POWER RUNS

Item Number	Process Quantities	Device Identification	Electronic Scram	Relay Scram	Backup Device Number	Backup Item Number
72[a]	Reactor Vessel Liquid Level Low	LS-1-6C		n - 1 foot	LS-1-16	73
73[a]	Reactor Vessel Liquid Level Low Backup	LS-1-16		n - 1 foot	LS-1-6C	72
74[a]	Primary Emergency Coolant Pumps Discharge Flow Low	FS-1-6B		5000 gpm	(FS-1-20B) (FS-1-21B)	75
75[a]	AC Primary Emergency (MIO) or DC Primary Emergency (MLI) Coolant Pump Recirculation Flow Low	(FS-1-20B) or (FS-1-21B)		175 gpm or 175 gpm	FS-1-6B	74
76	Reactor Inlet Water Temperature --- High	TRS-1-18C		130°F	TRS-1-11C, -12C, -13C, and -14C	77, 78 79, 80
77	Primary Coolant Quadrant #1 Outlet Temperature -- High	TRS-1-11C		135°F	DTRS-1-1D, TRS-1-18C	81, 76
78	Primary Coolant Quadrant #2 Outlet Temperature -- High	TRS-1-12C		135°F	DTRS-1-2D, TRS-1-18C	82, 76
79	Primary Coolant Quadrant #3 Outlet Temperature -- High	TRS-1-13C		135°F	DTRS-1-3D, TRS-1-18C	83, 76
80	Primary Coolant Quadrant #4 Outlet Temperature -- High	TRS-1-14C		135°F	DTRS-1-4D, TRS-1-18C	84, 76
81[a]	Quadrant #1 Temperature Rise High	DTRS-1-1D		5°F	TRS-1-11C	77
82[a]	Quadrant #2 Temperature Rise High	DTRS-1-2D		5°F	TRS-1-12C	78
83[a]	Quadrant #3 Temperature Rise High	DTRS-1-3D		5°F	TRS-1-13C	79
84[a]	Quadrant #4 Temperature Rise High	DTRS-1-4D		5°F	TRS-1-14C	80
85	High Neutron Level	Level Channel #7	1.2 N	1.2 N[b]	Level Channels 9, 8, and 6	86 87 88
86	High Neutron Level	Level Channel #9	1.2 N	1.2 N[b]	Level Channels 7, 8, and 6	85 87 88
87	High Neutron Level	Level Channel #8	1.2 N	1.2 N[b]	Level Channels 7, 9, and 6	85 86 88
88	High Neutron Level	Level Channel #6	1.2 N	1.2 N[b]	Level Channels 7, 9, and 8	85 86 87
89[a]	Test Plug Removed from either Log N Channel	TB-1 or TB-2			---	
90	None					

[a] Automatically cut in at power less than $3N_L$, and reactor inlet pressure as measured by PRS-1-11J less than 30 psig.

[b] Recalibrated to these operating levels by moving ion chambers.

TABLE VII-4-4

EXPERIMENT CONTROL AND SAFETY ACTIONS

Item Number	Parameters ^[a] 5 Loops 1 FW Loop	Device ^[a] Identification (Shown typically for Loop 2A)	Type of Power Reduction ^[b]				Backup Devices and Comments	Backup Item Number
			Setback ^[c]	Reverse	Fast Recovery Scram	Relay Scram		
61	Running Pump Failure	--				Yes		--
62	Loop Power Reduction Voltage, Low	--				Yes	Accomplished by under-voltage relay	
63	Primary Coolant Flow to In-Pile Tube, Low	FS-2A-1B	Yes	Yes	Yes	Yes	FS-2A-2B	64
64	Primary Coolant Flow from In-Pile Tube, Low	FS-2A-2B	Yes	Yes	Yes	Yes	FS-2A-1B	63
65	Primary Coolant Delta T across In-Pile Tube, High	DTS-2A-33B	Yes	Yes	Yes	Yes		
66	Primary Coolant Inlet Temperature to In-Pile Tube, High	TS-2A-41B	Yes	Yes	Yes	Yes	TS-2A-32B	67
67	Primary Coolant Outlet Temperature at In-Pile Tube, High	TS-2A-32B	Yes	Yes	Yes	Yes	TS-2A-41B	66
68	Primary Coolant Inlet Pressure to In-Pile Tube, Low	PS-2A-2B	Yes	Yes	Yes	Yes	PS-2A-4B	69
69	Loop Pressurizer Pressure, Low	PS-2A-4B	Yes	Yes	Yes	Yes	PS-2A-2B	68
70	First Half of Specimen Temperatures, High	TRS-2A-1B, etc (Test Dependent)	Yes	Yes	Yes	Yes	Second half of specimen temperature; any one temperature initiates power reduction.	71
71	Second Half of Specimen Temperatures, High	TRS-2A-2B, etc (Test Dependent)	Yes	Yes	Yes	Yes	First half of specimen temperature; any one temperature initiates power reduction.	70

[a] Identical for each of six loops except Cubicle Numbers 2A, 2B, 2C, 2D, 2E, 1D.

[b] Where more than one is given, one of them is preselected by test plugs.

[c] Fast or slow setback preselected by choice of connections in amphenol plug.

TABLE VII-4-5

MISCELLANEOUS CONTROL AND SAFETY ACTIONS

Item Number	Miscellaneous	Device Identification	Relay Scram	Backup Device Number	Backup Item Number
91	Reactor Instrument Air Pressure -- Low	PS-6-4B	60 psig	PS-6-4C	92
92	Reactor Instrument Air Pressure -- Low Backup	PS-6-4C	60 psig	PS-6-4B	91
93	Reactor Safety System Relay Supply Voltage -- Low	---		Inherently Releases Safety Magnets	
94	Commercial System Bus A Voltage Low	Phase A	0.42 N	Phase C	95
95	Commercial System Bus A Voltage Low Backup	Phase C	0.42 N	Phase A	94
96	Commercial System Bus B Voltage Low	Phase A	0.42 N	Phase C	97
97	Commercial System Bus B Voltage Low Backup	Phase C	0.42 N	Phase A	96
98	Diesel System Voltage Low	Phase A	0.42 N	Phase C	99
99	Diesel System Voltage Low Backup	Phase C	0.42 N	Phase A	98
100	Earthquake	Seismic Switch #1	Yes	Seismic Switch #2	101
101	Earthquake	Seismic Switch #2	Yes	Seismic Switch #1	100
102	Reactor Safety System Instrument Supply Voltage -- Low	27-1 through 27-8	Yes	---	---
103	Manual Operation	Pushbutton	Yes		
104	Test Plugs Removed from 3 out of 4 Level Channels	TB-3, TB-4, TB-5, and TB-6	Yes	---	
105	Test Plugs Removed from both Period Channels	TB-17 and TB-18	Yes	---	
106	Test Plugs Removed from Both Log N Channels	TB-1 and TB-2	Yes		
107	Test Plugs Removed from any One of the Above Process Variables and its Backup	---	Yes	---	
108	Any Dry Dock Test Plug Removed	SD-TB-1, SD-TB-2, or SD-TB-3	Yes	If Operating level is above 10^{-5} Np

FIG. VII-4-1 SAFETY SYSTEM SCHEMATIC.

5. SHIM INSERT DELAY TIMER.

The ATR, operating at full power, produces a high equilibrium concentration of xenon-135 parent in a relatively short time. Xenon buildup rate following a scram was calculated to be -0.125 percent Δk /minute and would require approximately 35 percent excess reactivity to override peak xenon poisoning. In order to override xenon poisoning, following a scram, it is therefore necessary to raise the power level rapidly.

A timer circuit, designed to delay the insertion of shim rods for 13 to 25 minutes following a scram (other than manual), has been incorporated in the ATR safety system as shown in Figure VII-4-2. This will permit the operator to recover from scrams that are cleared in a short time. Without this circuit, ATR scram recovery capability would be seriously downgraded because of the additional time required to withdraw both safety and shim rods.

The timer circuit is double tracked and is bypassed when the manual scram button is pushed or when the normal safety circuit bus power fails.

NOTE:

1. Relay R1-7 is energized when all safety rods are fully withdrawn.
2. Scram Relays RS-1 and RS-2 are energized during a scram.
3. Timer contacts IT-7A and IT-8A are closed when timer is energized.
4. Timer contacts IT-7B and IT-8B are closed after time period "x".
5. Timers will reset to zero when de-energized.
6. Relay RB-9 is energized and sealed-in by manual scram pushbutton.
7. Reverse relays RR-1 and RR-2 cause outer shim cylinders and neck shim rods to be inserted.

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FIG. VII-4-2 ATR SHIM INSERT DELAY TIMER.

5.1 Operation of Timer

The operation of the shim insert delay timer is as follows: (see Figure VII-4-2)

(1) After receiving a scram signal, contacts RS-1 and RS-2 will close to energize timers 1T-7 and 1T-8.

(2) When the safety rods leave their upper limit switches, relays R1-7A and 7B are de-energized closing contacts R1-7A and 7B.

(3) Timer contacts 1T-7A and 8A are closed as long as the timer is energized, and act as seal-in contacts through R1-7A and 7B contacts.

(4) If the scram signal has not cleared and reset, and the safety rods are not at their upper limit switches by the end of the timed period, reverse relays RR-1 and RR-2 are energized causing all shim rods and the selected regulating rod to insert.

(5) If during the timed period, the scram can be cleared and reset (RS-1 and 2 contacts open) and the safety rods withdrawn to their upper limit switches (relays 7A and 7B energized opening contacts R1-7A and 7B), the timer is de-energized.

(6) The timer will automatically reset to zero when de-energized.

(7) The operator can insert the shims and regulating rod at any time by pushing the manual scram button.

- (8) Loss of diesel-commercial power (normal power to the shim insert bus) will immediately insert all shim and regulating rods using battery-backed emergency power.
- (9) A reverse signal from a source other than the scram that energized the timer will cause the shim rods to insert immediately. A reverse signal occurs if the console key is turned off, automatically inserting all shim rods.
- (10) If the scram signal has been cleared and reset and the safety rods are being withdrawn when the elapsed time de-energizes the timer, the shim and regulating rods will insert at the same time the safety rods are being manually withdrawn.
- (11) Shim rod withdrawal is prohibited by interlocks if the safety rods are not at their upper limit switches.
- (12) If the reactor is not in the run regime, safety rod withdrawal is not permitted until the shim rods are completely inserted. Therefore, operating instructions will call for a manual scram if the reactor drops out of run regime with the timer energized.

5.2 Possible Accidents

Criticality considerations have been analyzed as related to the shim insertion delay timer.

5.21 Xenon Equilibrium Operation. If the reactor has operated at power for sufficient time to approach xenon equilibrium (approximately 3 percent Δk after 10 hours of full-power operation), a scram will result in xenon buildup at a rate of about 0.125 percent Δk /minute. The combined positive effects of regulating rod worth (drives out following a scram), fuel area moderator temperature, and fuel temperature increase are worth less than negative effects of xenon buildup during the time required to raise the safety rods. The reactor, therefore, cannot become critical on safety rod withdrawal.

5.22 Clean Core Operation. If the reactor has not approached xenon equilibrium (clean core) and a scram occurs, it is possible for the reactor to become critical during the latter stages of safety rod withdrawal (last several inches or less) because of the regulating rod worth and moderator and fuel temperature decrease. These positive reactivity effects are partially compensated by the negative reactivity caused by cooling the moderator in the reflector and flux trap regions. Although normal ATR operating practice does not permit reaching criticality with the safety rods not at their upper limit switches, the situation does not present a hazard since essentially the entire worth of the safety rods is available for scram action should this be necessary. In addition, since the need for rapid scram recovery is not necessary with a clean core, actual operating procedures will require manual partial shim insertion following a scram to prevent criticality occurring before the safety rods reach their upper limit switches.

It is, therefore, concluded that the operation of the shim insert delay timer does not present any additional operational hazards.

VII-5. PROCESS INSTRUMENTATION AND CONTROL

1. GENERAL

The process instrumentation measures, indicates, records, and controls process systems, sounds alarms on abnormal operation, and actuates reactor power reductions. Table VII-4-2 lists the corrective signals from the process instrumentation that initiate reactor power reductions.

All important process variables are recorded in the process and reactor control rooms. (See Figures VII-5-1, -5-2, and -5-3.) Measurements of minor importance are locally indicated.

All nuclear and process parameters which actuate reactor power reduction circuits are "double tracked" or "backed-up" to ensure reliable operation. "Double tracked" instrumentation utilizes independent protective circuits so that failure of any component in either power reduction circuit will not impair the proper operation of the other circuit. "Backed-up" instrumentation measures two independent process parameters with a predictable relationship. As a typical example, high reactor-inlet-water-temperature instrumentation is "backed-up" by quadrant outlet temperature instrumentation. Since high reactor-inlet-water temperatures will cause the quadrant-water-outlet temperature to rise, it will actuate high quadrant-outlet-temperature-power-reduction circuits.

2. PRIMARY COOLANT SYSTEM INSTRUMENTATION

2.1 General

The primary-coolant-system-instrumentation controls provide alarm signals and initiate reactor power reductions. This system is divided into the following categories:

- (1) Coolant temperature
- (2) Coolant pressure
- (3) Water power computation
(Flow and temperature rise)
- (4) Emergency coolant flow
- (5) Pump bearing temperatures
- (6) Seal water flow
- (7) Primary degassing tank
- (8) Bypass demineralizer
- (9) Coolant pH

MARK NO.	DEVICE IDENTIFICATION OR DESCRIPTION	MARK NO.	DEVICE IDENTIFICATION OR DESCRIPTION
51-1	FRS-1-1C, 1D, 1E & 1F	51-41	N16 POWER LEVEL SOUTHWEST INNER LOBE
51-2	FRS-1-2C, 2D, 2E & 2F	51-42	N16 POWER LEVEL NORTHWEST INNER LOBE
51-3	FRS-1-3C, 3D, 3E & 3F	51-43	N16 POWER LEVEL NORTH OUTER LOBE
51-4	FRS-1-4C, 4D, 4E & 4F	51-44	N16 POWER LEVEL EAST OUTER LOBE
51-5	DTRS-1-1B, 1C & 1D	51-45	N16 POWER LEVEL SOUTH OUTER LOBE
51-6	DTRS-1-2B, 2C & 2D	51-46	N16 POWER LEVEL WEST OUTER LOBE
51-7	DTRS-1-3B, 3C & 3D	51-47	LOG N CHANNEL NO. 4
51-8	DTRS-1-4B, 4C & 4D	51-48	LOG N CHANNEL NO. 5
51-9	TRS-1-11B & 11C	51-49	REACTOR PERIOD CHANNEL NO. 4
51-10	TRS-1-12B & 12C	51-50	REACTOR PERIOD CHANNEL NO. 5
51-11	TRS-1-13B & 13C	51-51	NEUTRON LEVEL CHANNEL NO. 7
51-12	TRS-1-14B & 14C	51-52	NEUTRON LEVEL CHANNEL NO. 9
51-13	TRS-1-15B & 15C	51-53	NEUTRON LEVEL CHANNEL NO. 8
51-14	PRS-1-11E, 11F, 11G & 11H	51-54	NEUTRON LEVEL CHANNEL NO. 6
51-15	FS-1-11C, 11D, 11E & 11F	51-55	COUNT RATE - CHANNEL NO. 1
51-16	FS-1-12C, 12D, 12E & 12F	51-56	COUNT RATE - CHANNEL NO. 2
51-17	FS-1-13C, 13D, 13E & 13F	51-57	SEISMIC SWITCH NO. 1
51-18	FS-1-14C, 14D, 14E & 14F	51-58	SEISMIC SWITCH NO. 2
51-19	FS-1-20A	51-59	UPPER LIMIT SWITCH SAFETY ROD SRNE
51-20	FS-1-21A	51-60	UPPER LIMIT SWITCH SAFETY ROD SRE
51-21	FS-1-6B	51-61	UPPER LIMIT SWITCH SAFETY ROD SRSE
51-22	FS-1-20B	51-62	UPPER LIMIT SWITCH SAFETY ROD SRS
51-23	PS-1-31E, 31F, 31G & 31H	51-63	UPPER LIMIT SWITCH SAFETY ROD SRSW
51-24	LS-1-1C, 1D, 1E & 1F	51-64	UPPER LIMIT SWITCH SAFETY ROD SRW
51-25	DRY DOCK - SAFETY RODS	51-65	UPPER LIMIT SWITCH SAFETY ROD SRNW
51-26	DRY DOCK - OUTER SHIMS	51-66	UPPER LIMIT SWITCH SAFETY ROD SRN
51-27	DRY DOCK - RECK SHIMS	51-67	RAPID RESPONSE FISSION BREAK DETECTOR (R-FB-2)
51-28	PS-6-4B	51-68	RAPID RESPONSE FISSION BREAK DETECTOR (R-FB-3)
51-29	PS-6-4C	51-69	FS-1-21B
51-30	FIS-1-23	51-70	COMMERCIAL SYSTEM LOW VOLTAGE BUS A-A-B
51-31	FIS-1-24	51-71	COMMERCIAL SYSTEM LOW VOLTAGE BUS A-B-C
51-32	LS-1-6C	51-72	COMMERCIAL SYSTEM LOW VOLTAGE BUS B-A-B
51-33	LS-1-16	51-73	COMMERCIAL SYSTEM LOW VOLTAGE BUS B-B-C
51-34	WRS-1-1B & 1C	51-74	DIESEL SYSTEM LOW VOLTAGE A-B
51-35	WRS-1-2B & 2C	51-75	DIESEL SYSTEM LOW VOLTAGE B-C
51-36	WRS-1-3B & 3C	51-76	LG 1 11C, 11D, 11E, & 11F
51-37	WRS-1-4B & 4C	51-77	DPS-1-21C & 22C
51-38	N16 POWER LEVEL NORTHEAST INNER LOBE	51-78	DPS-1-23C & 24C
51-39	N16 POWER LEVEL SOUTHEAST INNER LOBE		

MARK NO.	DESCRIPTION	MARK NO.	DESCRIPTION
1	ANNUNCIATOR	96	SELECTABLE READOUT 2-PEN RECORDER
2	PRIMARY COOLANT REACTOR INLET TEMPERATURE RECORDER	97	REACTOR COMPLEX TEMPERATURE RECORDER, 12 POINT
3	PRIMARY COOLANT REACTOR OUTLET TEMPERATURE RECORDER	98	HOLDING MAGNET CURRENT RECORDER, 16 POINT
4	R-ARY-1 STACK PARTICULATE ACTIVITY RECORDER	99	SIGMA BUS VOLTAGE RECORDER
5	R-ARY-2 STACK ALPHA ACTIVITY RECORDER	100	INSTRUMENT TEST BLOCKS (SEE SCHEDULE AT LEFT)
6	64 POSITION PATCH PANEL WITH 4 CONNECTORS - SEE DRAWING 670-W-6 DET. 9	101	QUADRANT 1 CORE DIFFERENTIAL PRESSURE RECORDER
7	PRIMARY COOLANT QUADRANT 1 OUTLET FLOW RECORDER	102	QUADRANT 2 CORE DIFFERENTIAL PRESSURE RECORDER
8	PRIMARY COOLANT QUADRANT 2 OUTLET FLOW RECORDER	103	QUADRANT 3 CORE DIFFERENTIAL PRESSURE RECORDER
9	PRIMARY COOLANT QUADRANT 3 OUTLET FLOW RECORDER	104	QUADRANT 4 CORE DIFFERENTIAL PRESSURE RECORDER
10	PRIMARY COOLANT QUADRANT 4 OUTLET FLOW RECORDER	105	PRIMARY COOLANT QUADRANT 1 DIFFERENTIAL TEMPERATURE RECORDER
11	PRIMARY COOLANT TOTAL OUTLET FLOW RECORDER	106	PRIMARY COOLANT QUADRANT 2 DIFFERENTIAL TEMPERATURE RECORDER
12	R-ARY-3 STACK GASEOUS ACTIVITY RECORDER	107	PRIMARY COOLANT QUADRANT 3 DIFFERENTIAL TEMPERATURE RECORDER
13	R-ARY-4 STACK FISSION GAS ACTIVITY RECORDER	108	PRIMARY COOLANT QUADRANT 4 DIFFERENTIAL TEMPERATURE RECORDER
14	NEUTRON LEVEL RECORDER - RETRANSMITTER CHANNEL NO. 7	109	PRIMARY COOLANT REACTOR DIFFERENTIAL TEMPERATURE RECORDER
15	NEUTRON LEVEL RECORDER - RETRANSMITTER CHANNEL NO. 9	110	PRIMARY COOLANT QUADRANT 1 OUTLET TEMPERATURE RECORDER
16	NEUTRON LEVEL RECORDER - RETRANSMITTER CHANNEL NO. 8	111	PRIMARY COOLANT QUADRANT 2 OUTLET TEMPERATURE RECORDER
17	REACTOR PERIOD RECORDER - RETRANSMITTER CHANNEL NO. 4	112	PRIMARY COOLANT QUADRANT 3 OUTLET TEMPERATURE RECORDER
18	REACTOR PERIOD RECORDER - RETRANSMITTER CHANNEL NO. 5	113	PRIMARY COOLANT QUADRANT 4 OUTLET TEMPERATURE RECORDER
19	WOR NO. 1 - POWER LEVEL SET - POINT WITH INDICATING LIGHTS	114	SELECTOR SWITCH FOR SAFETY ROD LOWER LIMIT AND SEAT SWITCH
20	WOR NO. 2 - POWER LEVEL SET - POINT WITH INDICATING LIGHTS	115	ATR LOCAL EVACUATION TELEPHONE TYPE KEY SWITCH
21	NEUTRON LEVEL RECORDER - RETRANSMITTER CHANNEL NO. 8	116	SITE EVACUATION AUTOMATIC ELECTRIC TYPE DA-41
22	NEUTRON LEVEL RECORDER - RETRANSMITTER CHANNEL NO. 6	117	POWER ON PILOT LIGHT
23	TOTAL WATER POWER RECORDER	118	LOOP DISCONNECT SWITCHES FOR LOOPS 2A, 2B, 2C, 2D, 2E, 1A, 1B, 1C, AND 1D RESPECTIVELY
24	LOG N RECORDER - RETRANSMITTER CHANNEL NO. 4	119	RADIATION MONITORING AND HAY RESET PUSH BUTTONS AND LIGHTS
25	LOG N RECORDER - RETRANSMITTER CHANNEL NO. 5	120	SAFETY SYSTEM BUS X TEST LAMP
26	QUADRANT 1 WATER POWER RECORDER	121	SAFETY SYSTEM BUS Y TEST LAMP
27	QUADRANT 2 WATER POWER RECORDER	122	SAFETY SYSTEM BUS H TEST LAMP
28	QUADRANT 3 WATER POWER RECORDER	123	SAFETY SYSTEM BUS TEST PUSHBUTTON
29	QUADRANT 4 WATER POWER RECORDER	124	CONTROL SYSTEM POWER SUPPLY GROUND DETECTOR - INDICATING LIGHTS
30	N16 LEVEL RECORDER - RETRANSMITTER - NORTHEAST INNER LOBE	125	RELAY CABINET POWER SUPPLY GROUND DETECTOR - INDICATING LIGHTS
31	N16 LEVEL RECORDER - RETRANSMITTER - SOUTHEAST INNER LOBE	126	AMPLIFIER CIRCUIT #1 GROUND DETECTOR - INDICATING LIGHTS
32	N16 LEVEL RECORDER - RETRANSMITTER - SOUTHWEST INNER LOBE	127	AMPLIFIER CIRCUIT #2 GROUND DETECTOR - INDICATING LIGHTS
33	N16 LEVEL RECORDER - RETRANSMITTER - NORTHWEST INNER LOBE	128	AMPLIFIER CIRCUIT #3 GROUND DETECTOR - INDICATING LIGHTS
34	N16 LEVEL RECORDER - RETRANSMITTER - CENTRAL REGION	129	LEVEL CHANNEL LOW RANGE - INDICATING LIGHT
35	N16 LEVEL RECORDER - RETRANSMITTER - CENTRAL REGION MID-PLANE	130	LEVEL CHANNEL NORMAL RANGE - INDICATING LIGHT
36	N16 LEVEL RECORDER - RETRANSMITTER - NORTH OUTER LOBE	131	RECORDERS DTRS - 1-1, 2, 3, 4 & 5 HIGH RANGE-INDICATING LIGHT
37	N16 LEVEL RECORDER - RETRANSMITTER - EAST OUTER LOBE	132	RECORDERS DTRS - 1-1, 2, 3, 4 & 5 LOW RANGE-INDICATING LIGHT
38	N16 LEVEL RECORDER - RETRANSMITTER - SOUTH OUTER LOBE	133	MISCELLANEOUS READOUT - 2 CHANNEL OSCILLOGRAPH
39	N16 LEVEL RECORDER - RETRANSMITTER - WEST OUTER LOBE	134	OUTER SHIM DRIVE INNER LIMIT LIGHTS
40	IODINE FISSION BREAK MONITOR RECORDER	135	BLOWN FUSE INDICATOR LIGHTS
41	DEGASSING TANK VENT MONITOR RECORDER	136	REACTOR ROOM SPARY ON PUSHBUTTON
42	LOG COUNT RATE RECORDER - RETRANSMITTER NO. 1	137	REACTOR ROOM SPRAY OFF PUSHBUTTON
43	LOG COUNT RATE RECORDER - RETRANSMITTER NO. 2	138	SAFETY SYSTEM BUS G TEST LAMP
44	PRIMARY COOLANT REACTOR DIFFERENTIAL PRESSURE RECORDER		
45	PRIMARY COOLANT REACTOR INLET PRESSURE RECORDER		
46	FISSION - BREAK DETECTOR (R-FB-2)		
47	FISSION - BREAK DETECTOR (R-FB-3)		
48	SELECTABLE READOUT 2-PEN RECORDER		

FIG. VII-5-1 REACTOR CONTROL PANEL ARRANGEMENT.

FIG. VII-5-3 PROCESS WATER PANEL ARRANGEMENT.

(10) Reactor shielding cooling air temperatures

(11) Reactor Vessel liquid level.

2.2 Primary Coolant Temperature Control System

The primary coolant reactor inlet temperature is controlled by varying the flow of the secondary coolant to the primary heat exchangers. The secondary coolant flow rate is regulated by a control valve, TCV-1-10, which is positioned by a three-element temperature control system as follows:

(1) Primary coolant reactor inlet temperature

(2) Primary coolant water power

(3) Secondary coolant water power.

This system has the inherent advantage of anticipating inlet temperature change, and adjusting secondary coolant flow for proper balance of power and temperature in both systems. The system is shown in Figure VII-5-4.

The outputs of the temperature recording controller, TRC-1-10, and the secondary power recording controller, WRC-2-1, govern the secondary system flow as required to maintain a constant inlet temperature in accordance with the instantaneous primary and secondary coolant water power balance. A computing relay, TC-1-10, permits manual calibration of the ratio of primary to secondary water power levels to compensate for such factors as the primary coolant pumps power input, or the fact that the water for reactor seals and bottom flushing does not enter at primary coolant water temperature. Such calibration reduces the system dead band.

The inlet temperature, measured by averaging three resistance bulbs, is converted to a pneumatic signal by a potentiometric transmitter which has a calibrated accuracy of better than ± 0.25 percent.

The three-mode temperature recorder controller, TRC-1-10, the computing relay, TC-1-10, and the secondary power controller, WRC-2-1, are mounted on the process panel. Control valve TCV-1-10 position is also indicated on the process control panel in Figure VI-2-1.

The pneumatic signals of primary and secondary water power from WT-1-5 and WT-2-1, respectively, are derived by multiplying the coolant flow by the temperature rise.

If the primary coolant reactor inlet temperature rises to 127°F, an alarm is annunciated on the process control panel. Temperature recorder TRS-1-18, mounted on the reactor control panel, also measures the inlet temperature, by averaging three resistance bulbs and actuating an alarm, and initiates setback and scram at successively higher levels. The inlet temperature measurement is backed up by the reactor quadrant outlet temperature recorders TRS-1-11, -12, -13, and -14 which also actuate alarms and initiate setback and scram.

In order to protect the secondary coolant pumps, which have minimum flow limitations, the secondary coolant control valve TCV-1-10 has a stop set to

FIG. VII-5-4 PRIMARY COOLANT TEMPERATURE CONTROL SYSTEM.

limit the minimum flow to 2000 gpm. The secondary coolant control valve design flow at 250 MW reactor power is 31,200 gpm. The primary coolant will therefore be cooled below 125°F when the valve reaches its minimum stop if the reactor power is below 15 to 30 megawatts, depending upon the season of the year. Also, if the temperature control system should malfunction so as to close the control valve, the minimum stop assures at least 2000 gpm for removing decay heat from the primary system. Upon loss of instrument air supply pressure to TCV-1-10, the control valve will remain in its last position. This

prevents reactivity additions caused by reducing the reactor inlet water temperature which would occur if the valve were arranged to fail open.

The temperature at the primary side outlet of each of the four heat exchangers is measured by averaging three resistance thermometer bulbs (a fourth is installed at each location as a spare) and is recorded on temperature recorder TR-1-6, located on the process control panel. This recorder has an accuracy of ± 0.25 percent and reads each temperature in turn with four seconds between points. The recorder will fail upscale on element or amplifier failure. Any difference in the outlet temperature of one of the exchangers will indicate possible tube plugging or surface fouling.

2.3 Pressure Control and Pressurizing System

The primary coolant system pressure controller, PRC-1-1, senses the pressure at the reactor vessel inlet and modulates control valve PCV-1-1, which balances primary coolant outflow to the degassing tank against system inflow from the primary pressurizing pumps, and also balances the in-leakage at various gland seals and the bottom-head flushing water to maintain a constant reactor inlet pressure of 355 psig.

The maximum gland seal inflow is approximately 70 gpm. The flow rate from the primary pressurizing pumps is preset by FRC-1-8 up to 300 gpm. Accordingly, the inflow to the primary coolant system is approximately 370 gpm; hence, pressure control valve PCV-1-1 will, under steady conditions, discharge 370 gpm to the degassing tank as measured and recorded on FR-1-25. If the primary system pressure at the primary coolant pump discharge should increase beyond 410 psig, then pressure recorder controller PRC-1-8 will override the flow controller FRC-1-8 and reduce the primary pressurizing pump flow.

The reactor vessel inlet pressure is recorded on the process control panel and the reactor control panel. Also, high- and low-pressure alarms are annunciated in the process and reactor control rooms.

On extreme low pressure, pressure switch PS-1-1J de-energizes the solenoid valve PJV-1-1A to close the pressure control valve PCV-1-1. On the loss of power to the diesel commercial bus, solenoid valve PJV-1-1A will lose power and close pressure control valve PCV-1-1. The above actions will hold the pressure in the primary system even if the pressurizing and gland seal flows stop.

The pressure control valve is supplied with a high-speed positioner, so that the valve will stroke from full-open to full-close in approximately four seconds. This control valve fails closed upon loss of instrument air supply pressure or control signal, thus maintaining the primary coolant system pressure.

In summary, upon loss of instrument air pressure, commercial diesel power, or the primary pressurizing pumps, pressure control valve PCV-1-1 will close to maintain primary system pressure.

On extremely high or low reactor vessel inlet pressure a reactor power reduction is automatically initiated. The reactor inlet pressure is double tracked.

The primary system surge tank can damp pressure surges because it contains the air cushion for the system. Two level ranges are recorded in the process control room. LR-1-1 records the normal operating range and LR-1-2, the entire height of the vessel. The full-height measurement is used during the initial filling operation and when depressurizing the vessel. High- and low-level alarms are annunciated on the process and reactor control panels. On extremely high and low primary surge tank level a reactor power reduction is automatically initiated.

Air is periodically added to the surge tank to make up for air which is dissolved in the primary coolant. This makeup air is obtained from the plant 250-psig air system. A double block and bleed control valve system, MCV-1-1A and 1B and MCV-1-2, was provided to ensure a tight shutoff of the 250-psig pressurizing air to prevent it from leaking into the surge tank. The admission of the pressurizing air is accomplished by remote manual loaders located on the process control panel. Depressurizing is accomplished by a manual loader located on the process control panel which positions two pneumatic control valves, MCV-1-3A and 3B, which are in series to ensure tight shutoff.

On extreme low level in the surge tank, level switches LS-1-1G and 11G de-energize the 250-psig air supply block and bleed control valve solenoids, MJV-1-1A and 1B and MJV-1-1-2, to shut off the 250-psig air supply. This control prevents the further addition of air to the surge tank to form an air bubble in the primary coolant circulating system if loss of water lowers the level in the surge tank. Also, the surge tank low-level switch LS-1-1B alarms to indicate an abnormal reduction in coolant inventory. In order to maintain primary coolant inventory, both solenoid valves FJV-1-8 and FJV-8-68 are de-energized, opening control valves FRCV-1-8 and FCV-8-68, thus supplying adequate demineralized water to the degassing tank for both pressurizing pumps operating in parallel to supply approximately 700 gpm makeup to the primary system.

2.4 Water Power Measurements

The ATR power is computed from thermal and flow measurements for each quadrant and also the total system as shown in Figures VII-5-5 and -5-6.

2.41 Quadrant Flow, Core Pressure Drop, and Temperature Measurements. The coolant flow rate in each of the four core quadrants is recorded on the linear flow recorders FRS-1-1, -2, -3, and -4 located on the reactor control panel. High and low alarms are annunciated in the reactor control room. Extremely high and low flows in any quadrant will initiate reactor power reductions automatically by either of two completely independent flow instrument channels. In addition, the reactor overall pressure drop is recorded and alarmed in both the reactor and process control rooms. Extremely high differential pressure across any quadrant will initiate power reduction circuits, and stop one of the three running primary coolant pumps. An interlock prevents starting a fourth pump if three are operating.

The reactor inlet and quadrant outlet temperatures are each measured by averaging three resistance bulb elements. The temperature rise in each quadrant is determined from the differential temperature measurement of the averaged outlet and reactor inlet temperatures. The quadrant differential temperatures are recorded in the reactor control room on DTRS-1-1, -2, -3, and -4. A high temperature rise alarm is annunciated in the reactor control room. Extreme high temperature rise initiates a reactor power reduction automatically. Backup of

FIG. VII-5-5 REACTOR POWER HEAT BALANCE SYSTEM.

FIG. VII-5-6 REACTOR POWER BLOCK DIAGRAM.

quadrant high differential temperature is provided by the quadrant outlet temperature recorders.

2.42 Quadrant Power. Each quadrant's linear flow signal is multiplied by its temperature rise to determine quadrant water power. The quadrant water power recorders with high- and low-power alarms are located on the reactor control panel. Extremely high water power in any quadrant initiates a reactor power reduction, providing additional backup to both the quadrant flow and quadrant temperature measuring systems.

2.43 Total Water Power Measurement. (See Figure VII-5-6) The total primary water power measurement is obtained by adding the four linear quadrant flows and multiplying by the separately measured overall reactor temperature rise. The total flow and total water power are recorded on both the reactor control panel and the process control panel. Alarms are annunciated in the process and reactor control rooms on high water power. Each loop control panel will also have recorders for quadrant power and total reactor thermal power.

2.5 Primary Emergency Coolant Pump Instrumentation

Each primary emergency coolant pump has a flow measuring orifice in its recirculation line. Upon low recirculation flow indicating running pump failure, and upon loss of either commercial or diesel power, the reactor will be scrammed, and simultaneously the standby primary emergency pump will be started automatically. The standby dc pump will be started whenever all primary pumps are shut off. During flux mapping, both pumps must be running. The recirculation flow rates and the discharge flow to the primary system are recorded on the process control panel by FR-1-20, FR-1-21, and FR-1-6, respectively.

2.6 Pump Bearing Temperatures

The four primary coolant pumps, the two primary emergency coolant pumps, the two primary pressurizing pumps, and the two warm seal booster pumps are each provided with two bearing temperature measurement thermocouples to detect overheating. All points are recorded on a multipoint temperature recorder, TRS-1-19, located on the process control panel. This recorder has an accuracy of ± 0.25 percent with four seconds delay between points, and reads upscale on thermocouple burnout or amplifier failure.

All 20 bearing temperature points are wired to one common, high temperature alarm, TA-1-19, set at 200°F. The annunciator alarm is also located on the process panel.

2.7 Gland Seal and Bottom-Head Flush System

Sealing water is supplied to the gland seal system by two primary warm seal water booster pumps discharging at 285 psig and two primary gland seal pumps normally discharging at 410 psig. The primary warm-seal-water-booster-pumps discharge header and primary gland-seal-pumps discharge header are interconnected via a hand operated isolating valve.

The primary warm-seal-water booster pumps normally supply water to all of the low-pressure seal systems.

The primary gland seal pumps normally supply water to all the high-pressure seal systems, N-16 monitoring system and, when necessary, can also be used to supply water to the low-pressure seal systems.

2.71 Low-Pressure Seal System. The low-pressure seal system provides sealing water to the following reactor seals:

Neck shim drive housing seals (24)

Bottom-head pressure tube seals (9)

Safety rod drive seals (8)

Safety rod snubber tube flush (8)

Reactor bottom flush.

Under normal operating conditions only one primary warm-seal-water booster pump is operated. The second pump is a standby which is automatically started on low differential pressure between pump discharge pressure and reactor outlet pressure. Automatic starting of the idle pump is accomplished by two differential pressure switches located on the process control panel. In case of failure of both primary warm-seal-water booster pumps, the primary gland seal pumps will provide seal water for the low-pressure seal system. This is accomplished by opening the isolating valve in the tie-in piping between low- and high-pressure system.

The differential pressure control valve, located in the pump discharge, regulates the seal water pressure to 35 psig above reactor outlet pressure. The flow to the gland seals of the low-pressure seal system is measured and also indicated on individual gland seal rotameters with a high flow alarm annunciated on the process control panel. A leaky seal can be detected by observing the seal water flow rotameters. A continuous flow of water to the reactor vessel bottom head is provided to prevent settling of crud on the inside bottom of the vessel. This flow is manually controlled and measured by an indicating rotameter FIS-1-22 with a low-flow alarm contact that will actuate an annunciator on the process control panel.

2.72 High-Pressure Seal System. The high-pressure seal system provides water to the following:

Primary coolant pump seals

Primary emergency coolant pump seals

Primary warm-seal-water booster pumps,

HDW circulation pump seals

Outer shim drum drive seals (8)

Top-head pressure tube seals (9)

Reflector cylinder position indicators (16)

N-16 Monitoring system. (10)

Under normal operating conditions, only one primary gland seal pump is operated. The second pump is a standby which is automatically started on low discharge header pressure. Automatic starting of the idle pump is accomplished by two pressure switches on the pump discharge header. The differential pressure control valve, located in the reactor seal water line, regulates the seal water pressure to 25 psig above reactor inlet pressure. The self operated pressure reducing valve, located in the seal water line to the primary coolant pumps, primary emergency coolant pumps, primary warm-seal-water booster pumps, and HDW circulating pump seals reduces pump discharge pressure to 245 psig. The pressure control valve in the line to N-16 monitoring system provides the system with water at 395 psig. The flow to the gland seals of the high-pressure seal system and the flow to N-16 monitoring system is measured and also indicated on individual gland seal and N-16 monitoring rotameters. High-flow alarms for the high-pressure seal system are annunciated on the process control panel. N-16 monitoring system high- and low-flow alarms are annunciated on the reactor and the process control panels.

Provisions are made in the low- and high-pressure seal systems to connect a portable rotameter to each seal supply line to check and adjust the individual flow rates during operation.

2.8 Primary Degassing Tank

Instrumentation controlling the primary degassing tank is as follows:

(1) Recording controller, LRC-1-3, operates the inlet makeup control valve, LCV-1-3A, and the outlet drain control valve, LCV-1-3B, each of which is furnished with a split-range positioner. Normally, the inlet makeup valve is closed and the outlet valve maintains the liquid level by discharging excess coolant resulting from gland seal inflow to waste. The makeup control valve, LCV-1-3A, opens and the drain control valve, LCV-1-3B, closes on loss of instrument air to ensure an adequate water supply to the pressurizing pumps.

(2) The tank is also provided with an overflow line to ensure against the coolant entering the ventilation system in the event of high level. During certain transients, the outlet drain valve will be closed and the makeup valve will be opened to increase the water inventory and to ensure that the primary pressurizing pumps have adequate suction head.

(3) Two positive-acting ball float type level switches actuate alarms on high or low level in the tank. High level is 12 inches above normal; low, 12 inches below. The annunciator alarms are located on the process and reactor control panels.

(4) The pressure in the tank is indicated on the process control panel. Normally, a slight vacuum of three inches of water gauge is maintained by blower 670-HVE-18.

(5) The drain flow rate through valve LCV-1-3B is recorded on FR-1-7 on the process control panel.

(6) The flow rate of the air exhausted by the blower is adjusted by using a portable pitot tube to measure the flow while the damper is set to the correct position manually.

2.9 Bypass Demineralizer Instrumentation -- Normal Operation

The flow rates through the two cation and two anion units are controlled by individual indicating manual loading stations on the process control panel, which indicate the flow rate and permit positioning the respective flow control valves. In addition, there are two other manual loading stations on the process control panel, each respectively controlling a valve which can bypass the flow around the cation units, the anion units, or, by working together, both units.

In the event of high primary coolant inlet temperature, alarm TA-1-10A is sounded at 133°F on the process control panel. If the temperature continues to rise the operator will shut flow control valves FCV-3-1, -2, -3, and -4 to protect the resins against damage from high temperature.

An indication of the condition and performance of each bed can be obtained from the differential pressure reading across each of the two cation and two anion units. Readings are indicated on the process panel by receiver gauges. Another differential pressure transmitter is located across the effluent filter. This gauge gives an indication on the process control panel of the buildup of solids on the filter from the demineralizer units.

On the unit effluent line (return to primary system) are:

- (1) A pH recorder with high and low alarms
- (2) A conductivity recorder with high alarm
- (3) A flow recorder.

These three recorders and the alarms are mounted on the process control panel. A high pH will indicate an exhausted or malfunctioning cation bed; a low pH, an exhausted anion bed. High conductivity will show a breakthrough of electrolytes from either bed.

2.91 Regeneration. The regeneration cycle is initiated manually, and controlled automatically, by means of a program timer. The program timer sets the position of the program valves at each step in the regeneration cycle.

Regeneration interlocks are as follows:

- (1) High temperature of caustic regenerant solution will cut off steam supply to heat exchanger.
- (2) High conductivity at the outlet of the anion resin unit being regenerated will keep resin transfer valve (ECV-3-8) closed.

(3) Loss of caustic dilution water will prevent the strong caustic pump from starting if the regeneration cycle is actuated.

2.10 pH Control Primary and Secondary Coolant Systems

The pH of each coolant system is separately controlled by means of equipment consisting of a pH sampling and detecting element, a recorder-controller, and a "Statotrol" device which controls the varying speed of a metering pump, thus proportioning the acid feed into the respective systems. On the primary system, a continuous sample is taken from the inlet line to the reactor, and acid for pH adjustment is injected into the outlet line from the reactor vessel. On the secondary system, a continuous sample is taken on the inlet line to the primary heat exchangers, and the acid is injected into the cooling tower basin. The recorder-controller in each case will be equipped with terminals for a 10 to 50-mV signal to the data system. The range of both instruments is pH 2-12 with a system accuracy of $\pm$ one percent of full scale. Both pH controllers are furnished with two SPDT adjustable contacts for high and low pH alarms.

2.11 Reactor Shielding Temperatures

The temperature of the high density and structural concrete surrounding the reactor is measured in each of the four quadrants. A thermocouple is provided in each quadrant at the core midplane in both layers of concrete, for a total of eight. In addition, there are four thermocouples, 90 degrees apart, in the annulus between the reactor and high density concrete, which, on high temperature, indicate loss of cooling air flow.

These 12 points are recorded on a multipoint temperature recorder TRS-1-28, located on the reactor control panel. This recorder will have an accuracy of $\pm$ 0.25 percent with four seconds between points, and will fail upscale on thermocouple burnout or amplifier failure.

The four points in the high density concrete are wired to a common high temperature alarm switch; the four in the structural concrete, to a second common high temperature alarm switch; the four in the air flow, to a third high temperature alarm switch.

The annunciator alarms are also located on the reactor control panel.

Six thermocouples, TE-1-29-1 through -6, are installed in the working canal walls for checking concrete temperatures resulting from storage of freshly discharged fuel elements. The measurements are read locally with a portable instrument.

2.12 Reactor Vessel Liquid Level Instrumentation

During low power flux mapping the reactor vessel liquid level is at the bottom of the reactor top head. The vessel liquid level is recorded on LR-1-6 on the process control panel. High and low liquid level alarms are annunciated in both the reactor and process control rooms. If the liquid level drops to one foot below normal liquid level, a scram is initiated by switch LS-1-6C. The low level is double tracked by switch LA-1-16. The range of the vessel liquid level recorder is from the lower overflow nozzle to top of the vessel.

If the water level in the reactor vessel falls to the level of the reactor outlet lines, the hydraulically operated firewater shutoff valves LCV-1-7A and 7B

will open to supply emergency fire water to the reactor core. The fire-water shutoff valves will open after a scram has been initiated only if the augmented primary water inflow supply has failed to keep up with an uncontrolled loss of primary coolant.

VII-6. RADIATION MONITORING SYSTEM (RMS)

1. GENERAL

To achieve adequate monitoring for the ATR and the surrounding areas, the radiation monitoring system design includes the following key features:

- (1) Placement of the monitors at optimum locations to ensure full coverage against conceivable accidents at all the locations operators are required to visit.
- (2) Utilization of signals from the radiation monitoring system to automatically isolate individual area ventilation systems to limit the spread of airborne activity. All radiation monitoring interlocks may, however, be bypassed at the emergency diesel engine start-up panel (ESP) in the diesel pit to allow an operator to ventilate the building when conditions permit.
- (3) Alarms at all fixed monitors are duplicated at two separate control locations: The Health Physics Panel and the Reactor Control Panel. Further, that portion of the radiation monitoring system designated as process monitors (seven area monitoring channels) is indicated and annunciated on the process control panel.
- (4) Monitoring of both direct radiation and airborne activity. This duplication obviates the need to assume that one form of activity will be accompanied by a sufficient amount of the other to actuate either type detector.
- (5) The automatic throw-over diesel-commercial system supplies power for radiation monitoring equipment connected to the Health Physics Panel and its interlocks with H and V equipment.
- (6) Portable monitoring for beta, gamma, and alpha activity is used for specific areas as required by operators.

The following types of monitoring equipment are provided:

- (1) 36 Beta-Gamma Monitors
- (2) 23 Airborne Particulate Monitors
- (3) 3 Hand and Foot Monitors
- (4) 17 Portal Monitors
- (5) One Analytical Gas Monitor
- (6) One Iodine Type Ruptured Fuel Element Monitor.

In addition, portable monitors will be available from the large inventory of instruments at the MTR-ETR area. The system described in the following section will be permanently installed.

2. BETA-GAMMA MONITORS

A total of 36 channels of beta-gamma monitoring equipment will be provided. Twenty-nine of these channels will be used for area monitoring and seven for process monitoring. The location of each channel is given in Table VII-6-1.

TABLE VII-6-1

LOCATION OF BETA-GAMMA MONITORS

<u>1. Location of Beta-Gamma Area Monitors</u>		
<u>Tag Number</u>	<u>Location</u>	<u>Channel Range</u>
RE-GM-1	Hallway of Office Building Mezzanine	0.1 mr/hr -- 100 r/hr
RE-GM-2	Hallway of Critical Facility Mezzanine	0.1 mr/hr -- 100 r/hr
RE-GM-3	Above Reactor Head	0.1 mr/hr -- 100 r/hr
RE-GM-4	On Reactor Handrail	0.1 mr/hr -- 100 r/hr
RE-GM-5	Critical Facility	0.1 mr/hr -- 100 r/hr
RE-GM-6	Control Room	0.1 mr/hr -- 100 r/hr
RE-GM-7	Utility Area (Main Floor)	0.1 mr/hr -- 100 r/hr
RE-GM-8	East End of Storage Canal	0.1 mr/hr -- 100 r/hr
RE-GM-9	West End of Storage Canal	0.1 mr/hr -- 100 r/hr
RE-GM-10	Middle of Storage Canal	0.1 mr/hr -- 100 r/hr
RE-GM-11	Maintenance Area	0.1 mr/hr -- 100 r/hr
RE-GM-12	West Wall of Reactor Room	0.1 mr/hr -- 100 r/hr
RE-GM-13	East Wall of Reactor Room	0.1 mr/hr -- 100 r/hr
RE-GM-14	East of Cubicles (1st Basement)	0.1 mr/hr -- 100 r/hr
RE-GM-15	Southeast of Cubicles (1st Basement)	0.1 mr/hr -- 100 r/hr
RE-GM-16	Southwest of Cubicles (1st Basement)	0.1 mr/hr -- 100 r/hr
RE-GM-17	Northwest of Cubicles (1st Basement)	0.1 mr/hr -- 100 r/hr

TABLE VII-6-1 (cont.)

LOCATION OF BETA-GAMMA MONITORS

Tag Number	Location	Channel Range
RE-GM-18	Opposite Door to Degasifier and Heat Exchanger Vault	0.1 mr/hr -- 100 r/hr
RE-GM-19	Utility Area (1st Basement)	0.1 mr/hr -- 100 r/hr
RE-GM-20	At Canal Viewing Window	0.1 mr/hr -- 100 r/hr
RE-GM-21	Northeast of Cubicles (2nd Basement)	0.1 mr/hr -- 100 r/hr
RE-GM-22	Southeast of Cubicles (2nd Basement)	0.1 mr/hr -- 100 r/hr
RE-GM-23	Southwest of Cubicles (2nd Basement)	0.1 mr/hr -- 100 r/hr
RE-GM-24	Northwest of Cubicles (2nd Basement)	0.1 mr/hr -- 100 r/hr
RE-GM-25	Outside Ion Exchanger Vault	0.1 mr/hr -- 100 r/hr
RE-GM-26	Entrance Tunnel to Subpile Room	0.1 mr/hr -- 100 r/hr
RE-GM-27	Subpile Room	0.1 mr/hr -- 100 r/hr
RE-GM-28	Rod Access Room	0.1 mr/hr -- 100 r/hr
RE-GM-29	Motor Room	0.1 mr/hr -- 100 r/hr

2. Location of Beta-Gamma Process Monitors

Tag Number	Location	Channel Range
RE-GM-30	Outlet of Heat Exchanger #1 (670-M-2)	0.01 mr/hr -- 10 r/hr
RE-GM-31	Outlet of Heat Exchanger #2 (670-M-3)	0.01 mr/hr -- 10 r/hr
RE-GM-32	Outlet of Heat Exchanger #3 (670-M-4)	0.01 mr/hr -- 10 r/hr
RE-GM-33	Outlet of Heat Exchanger #4 (670-M-5)	0.01 mr/hr -- 10 r/hr
RE-GM-34	In the Stack's Breech	0.1 mr/hr -- 100 r/hr
RE-GM-35	On Exhaust Fan HV-17's Plenum	0.1 mr/hr -- 100 r/hr
RE-GM-36	On Degassing Tank Vent	0.1 mr/hr -- 100 r/hr

A six-decade logarithmic Geiger-Muller detecting system installed on the wall of each of the above locations is provided. The location was selected so that the solid angle subtended by the detector is not restricted by equipment. All area monitors are equipped with a local meter and alarm. The process channels have indicating meters and alarms located on the process control panel. All detectors have outputs recorded and annunciated in the Health Physics Office. All alarms are also annunciated in the Reactor Control Room. The detectors for the primary heat exchangers are six-decade (0.01 mr per hour to 10 r per hour) halogen self-quenching Geiger-Muller detectors. The remaining 32 detectors are of the same type; however, the six-decade range specified is 0.1 mr per hour to 100 r per hour. The output of each channel is recorded on a bank of twelve strip chart recorders which, together with the power supplies and station indicators, are located in the Health Physics Office.

3. AIRBORNE PARTICULATE MONITORING SYSTEM

Twenty-three cart mounted, interchangeable fixed filter, air particle monitors are provided. Six carts (R-CAM-1 through -6) are mobile units that may be positioned at any location in the plant. The remaining 17 will be located as follows:

<u>Tag Number</u>	<u>Location</u>
<u>1. First Floor</u>	
R-CAM-7	Reactor Working Canal
R-CAM-8	West End Canal Area
R-CAM-9	East End Canal Area
R-CAM-10	Critical Facility
R-CAM-11	Reactor Control Room
R-CAM-12	Hallway to Health Physics Field Office
R-CAM-13	Sample Preparation Laboratory
R-CAM-14	Reactor Area
<u>2. First Basement</u>	
R-CAM-15	East Side
R-CAM-16	South Side
R-CAM-17	West Side
<u>3. Second Basement</u>	
R-CAM-18	South Side (Near subpile room door)
R-CAM-19	East Side

<u>Tag Number</u>	<u>Location</u>
R-CAM-20	Southwest Side
R-CAM-21	Northwest Side (at fan room door)
4. <u>Process Area (First Floor)</u>	
R-CAM-22	General Storage and Laydown Area
R-CAM-23	Process Control Room

The 17 "fixed location" carts are provided with 10-foot power supply and signal transmission cables to permit Health Physics personnel to optimize their locations. All carts provide visual and audible alarms locally. The six unassigned carts record their output on strip-chart recorders mounted on the carts. The 17 "fixed" carts are equipped with jacks and panel cutouts for similar recorders which are normally not needed because the "fixed" carts transmit their outputs to the Health Physics Office where 9 two-pen strip chart recorders provide a continuous record and annunciate high and intermediate level alarms for the system. Three recorders for optional use with any of the 17 "fixed" carts were provided. The alarms are also annunciated in the Reactor Control Room. The units employ a Geiger-Muller detector and a Linear Automatic Range Change ratemeter. The pump motor may be controlled from the Health Physics panel. This feature permits the decay of material held on the filter to be followed, thus aiding identification. Ranges are:

Low	0 to 2000 cpm
Intermediate	0 to 10,000 cpm
High	0 to 20,000 cpm.

All carts take their power from suitable diesel-commercial ac power system with twist lock connectors to prevent accidental detachment.

4. HAND AND FOOT MONITORS

Three hand and foot monitors are provided at the following locations:

Corridor of shift supervisor's office

Corridor of critical facility

Corridor near utility supervisor's office.

The units have two counters for each hand and two for each foot and one counter for an external probe for monitoring hair and clothing. Each of these nine Geiger-Muller counters have four ranges (0 to 500 cpm, 0 to 2000 cpm, 0 to 5000 cpm, and 0 to 20,000 cpm) for hands, the feet, and the probe. Five separate meters are provided for each unit: one for each foot, one for each hand, and one for the probe counter. The meters are mounted on the cabinet. Sensitivity

will be such that a 0.2 MeV beta particle will be detected. Audible and visual alarm setpoints are adjustable over each range.

5. PORTAL MONITORS

There are 17 portal monitors installed at the following locations:

Doorway D-12	Entry Health Physics Issue Room
Doorway D-14	South Reactor Area
Doorway D-16	Critical Facility Area
Doorway D-18	West Storage Canal Area
Doorway D-20	North Storage and Laydown Area
Doorway D-22	North Utility Area
Doorway D-25	East Storage Canal Area
Doorway D-28	Office Building Corridor
Doorway D-30	Office Entrance
Doorway D-49	West Reactor Area
Doorway D-50	Northwest Reactor Area
Doorway D-52	Northeast Reactor Area
Doorway D-54	East Storage Canal Area
Doorway D-67	West Storage Canal Area
Doorway D-71	Critical Facility Area
Head of Stairway Number 3	Utility Storage Area
Doorway D-101	Office Building Mezzanine

These positions were selected so that all personnel would pass through a portal monitor whenever leaving a potentially radioactive area.

Each unit contains 11 Geiger-Muller detectors located in the floor, sides, and above the entryway.

The detectors will be sensitive enough to detect a 0.2 MeV beta particle and will initiate a local audible and visual alarm on high level.

6. ANALYTICAL GAS MONITOR

This system resolves the radioactivity carried by the effluent gas being discharged to the stack into four components. These components are:

- (1) Particulate beta-gamma
- (2) Alpha particulate (after a six-hour delay to permit the decay of naturally occurring isotopes)
- (3) Gaseous beta-gamma
- (4) Fission daughter.

The equipment to monitor these activities is located in and outside the fan room in the northwest corner of the second basement. The gas sample is taken by an isokinetic sample nozzle located in the underground portion of the stack duct. The sample is drawn through a moving filter mechanism which collects all of the particulate material. The activity is monitored by a scintillation detector for beta-gamma radiation. The tape is monitored for alpha activity six hours later by a scintillation detector. (Six hours allows sufficient time for the combined daughters of the natural radon and thoron activity to decay to negligible value.) After filtration, the gas is drawn into an annular pig where the gaseous activity is measured by a scintillation detector. The gas is then allowed to pass through a delay line sized to provide a two-minute delay at a flow of 10 cfm. The delay allows a portion of the noble gas activities present to decay and form particulate daughters. These daughters are filtered and monitored by a moving filter-detector similar to the first unit described above. Since the daughters of the fission product noble gases are radioactive while the daughter of Argon-41 is not, this signal permits determination of the portion of the gaseous activity measured previously that was due to fission products. The gas is returned to the ventilation duct and discharged.

The output of all four scintillation detectors is transmitted to recorders on the Health Physics Panels. Adjustable alarms are provided by the logarithmic, transistorized ratemeters, which operate annunciators in both the Health Physics and Reactor Control Rooms. Additional alarms and monitoring component functions of this system are also included on the Health Physics Panel.

7. IODINE TYPE RUPTURED FUEL ELEMENT MONITOR

This monitoring system takes a metered flow of primary coolant from the outlet line of the primary coolant, purification system, cation demineralizer, concentrates iodine and other anions on a demineralizer column, and employs a gamma-ray detector to monitor the concentrated iodine activity to indicate the release of fission products. The equipment is located in the waste tank motor area below the second basement level. The sample is discharged to the warm waste tank directly below the motor area. The detector is operated with a discriminator setting of about 0.8 MeV to bias out activity from Cr-51 and Mo-99

radicals which concentrate on the anion column. The higher energy radiations are predominantly due to iodine activity. The scintillation detector is sensitive enough to give an 11,000-cpm signal for each microcurie per cc of I-131 in the anion column. Two check sources, Co-60 (high energy) and Cs-137 (low energy), both of 9.0 microcuries, were provided to permit the operator to test the setting of the discriminator. Lead shields the detector from the anion column during calibration. The distance between the detector and column can be varied by the operator to keep the instrument on the proper scale. These readings are recorded and annunciated in both the Health Physics Office and Reactor Control Panel. In addition, low flow to the instrument is annunciated in the Health Physics Office and the Reactor Control Room.

8. INTERLOCK RELAYS FOR H AND V SYSTEM

Four relays were furnished on the Health Physics Monitoring Panel for the purpose of actuating interlocks with the Heating and Ventilating System.

Interlock Relay Number 1 function:

- (1) Close the stack breeching damper
- (2) Shut off the main exhaust fans HVE-17A and -17B
- (3) Shut off the following equipment associated with the reactor area supply systems:

HVS-1 and associated butterfly dampers

HVR-1 Butterfly Damper

HVS-4 Two Butterfly Dampers

HVS-5 Two Butterfly Dampers

HVE-9 and one Butterfly Damper

HVA-2 and three associated Butterfly Dampers

HVE-4.

Interlock Relay Number 2 independently shuts off reactor supply systems as specified for Interlock Relay Number 1 (3) above and diverts the exhaust of HVE-1 to the main exhaust plenum by means of two dampers.

Interlock Relay Number 3 shuts off the critical facility supply system HVS-2 and its associated dampers.

Interlock Relay Number 4 shuts off the storage canal ventilation system (HVS-3) and closes the associated motor operated dampers and roof ventilators.

The four relays are energized below the preset trip points of the actuating channels and open to perform the interlock actions. Each relay is provided with a contact to actuate two indicating lights, one on the Health Physics Panel and one on the Reactor Control Panel. Lights are on while the relay is open. Four manual reset pushbuttons and four indicating lights are provided on Reactor Control R-9. These interlocks will reactivate if reset while an actuating condition continues. However, the interlocks can be positively overridden by means of a pushbutton at the emergency diesel start-up panel.

Interlock Relay Number 1 is actuated by either the stack breech monitor (GM 34) or the exhaust plenum monitor (GM 35). The exhaust plenum monitor incorporates an adjustable time delay in its interlock output circuit to allow for transit time of the gas in the 300-foot duct to the stack.

Interlock Relay Number 2 is actuated by any one of the following inputs:

- (1) A coincidence of the interlock signals of CAM 7 and CAM 14 located in the reactor room
- (2) A two out of three coincidence of GM 2, GM 5, and GM 6 which surround the reactor
- (3) A two out of three coincidence of GM 23, GM 24, and GM 25 located in the Second Basement.

Interlock Relay Number 3 is actuated by CAM-10 -- Critical Facility Airborne Particulate Monitor.

Interlock Relay Number 4 is actuated by either of the two CAM's monitoring the canal: CAM 8 or CAM 9.

All input signals to the interlock relays that do not require a coincidence for actuation are provided with a bypass switch on the Health Physics Panel (GM 34 and 35; CAM 8, 9, and 10). This switch permits the operator to calibrate and test the actuation of these channels without shutting off the ventilation system. Use of these bypass switches is under administrative control.

9. EXPERIMENTAL CUBICLE MONITORING

9.1 Area

Cubicle monitoring prior to entry is accomplished by inserting a Geiger-Muller detector through the cubicle wall after withdrawing a normally closed plug.

9.2 Process

Monitoring instruments are installed with each experiment loop. Loop instrument selection is dependent upon the experimental program. The one common design criterion for the experimenters' monitoring instruments is that they must be able to detect a release of fission products from the experiment to the loop coolant.

Each of the six pressurized water loops has two gross beta-gamma channels (six decade) for monitoring the following:

- (1) The piping to the purification system
- (2) The piping from the reactor just inside the cubicle.

For each of the above channels, a recorder with two adjustable switch contacts is located on the experimenter's control panel. The first switch contact actuates a high radiation alarm and the second switch is a spare contact for a future reactor power reduction.

A suitable monitoring system will be designed for the gas-cooled loop to meet the same objectives.

VII-7. DATA PROCESSING SYSTEM

A data processing system is provided which is capable of performing scanning, logic check, memory storage, and computations.

1. NORMAL SCANNING

The system can scan 300 analog inputs at the rate of 60 per second. The experimental facilities are assigned 220 points. The scanning cycle is utilized mainly for operational considerations and accomplishes the following:

1.1 Comparison of Analog Inputs Against Limits

This scanning operation compares each of the 300 input signals with pre-determined limits to inform the operator of each abnormal condition, the absolute value of the abnormal quantity, and the duration of the abnormality. This operation is in addition to normal indication, recording, and annunciation from analog instrumentation at the process or the reactor control panels.

1.2 Routine Periodic Logging

All logged readings are taken within a five-second scan.

1.3 Rapid Computation of Predetermined Quantities

Instantaneous computations of interest to the operator are performed as requested to provide simplified information concerning total plant operation.

1.4 Comparison of Computed Quantities Against Limits

Each computed quantity is compared with predetermined limits to inform the operator of abnormal conditions by alarm and print-out.

2. INDEPENDENT LOGIC CHECK OF THE OPERATIONS OF THE CONTROL SYSTEM RELAY NETWORK

The data processing system continuously performs an independent check of the control system relay network which controls the various regimes and rod withdrawal permissives. This system independently scans the permissive condition circuits and inputs each time the operator initiates a rod withdrawal or changes the operating regime. This scan looks at the binary inputs directly and at the status of the analog inputs as of the latest scan cycle. The system alarms if the operator attempts to perform an illogical operation. Attempts to perform operations which are not allowed by the safety system are also logged. All power reductions are logged with time of occurrence. In addition, each manual action of withdrawal or insertion is logged in actual time.

Thus, the Data Processing System serves as a backup to the operational interlocks. It has further value in that it can point out repetition of potentially unsafe operating practices. It also has the ability of simultaneously following a

number of abnormal conditions whereas an operator would tend to concentrate on only one.

3. MEMORY

The system provides a memory of the past five minutes at all times. Following an automatic power reduction, no new memory is retained and the existing memory record is recorded. This is most useful for later analysis of the conditions leading to a reactor power reduction or shutdown.

4. ACCELERATED SCANNING DURING TRANSIENTS

The data system provides accelerated scanning of preselected quantities during any transient period of operation. The initiation of an automatic power reduction causes the computer to shift to an accelerated scanning of preselected inputs of special significance during transient behavior. It is anticipated that one-fifth of the total number of analog inputs will be of special "transient" significance. This permits an increase of five-to-one in the scanning rate of the significant inputs.

5. CALCULATION AND INTERPRETATION OF INFORMATION

Based upon the nuclear and process inputs, the system computes and indicates the following:

- (1) Quadrant and total reactor power
- (2) Integration of quadrant and total power
- (3) Breakdown of total reactor power into central region and four lobe regions
- (4) Mass flow of stack gas
- (5) Radioactive stack gas release
- (6) Integrated radioactive stack gas release
- (7) Xenon inventory
- (8) Xenon transient poisoning override time
- (9) Surface heat flux in each experiment
- (10) Experiment loop thermal power.

The system is required to compute the five lobe power balance (item 3 above) because the maintenance of precise and constant power distribution pattern is of prime importance in fulfilling the requirements of the experiments in the reactor. The maintenance of the desired power distribution will require the rapid analysis of operating data and the readjustment of the reactor shims. The frequency with which the readjustments are required and the large quantity of data to be analyzed would require a large portion of the operator's time if the computer were not available.

VII-8. ATR SYSTEM TRANSIENT BEHAVIOR

ATR performance was investigated by a closed loop analog simulation of the various reactor and process control systems influencing reactor power.

The primary objectives were to determine:

- (1) Plant response during normal power level maneuvers
- (2) Plant response during specified equipment malfunctions
- (3) Operational limitations imposed by equipment design
- (4) Response to outside temperature variations.

An experimental program to study the ATR servo regulator rod characteristics and compatibility with reactor kinetics was also performed.

1. CLOSED LOOP ANALOG STUDY OF THE ATR

The ATR reactor performance was investigated using a closed loop analog simulation of the several control systems effecting reactor power. The simulation was based on the functional diagram shown in Figure VII-8-1.

Equations simulating the following systems were derived:

- (1) Reactor Kinetics -- point-reactor kinetic model
- (2) Reactor heat transfer -- uniform core heat generation using ATR fuel and channel composition and geometry
- (3) Reactor Control System -- power demand-time curve for fast and slow start-up and setback
- (4) Regulating rod drive system -- second-order velocity servo system with velocity feedback
- (5) Reactor Automatic Power Reduction -- ATR setbacks, reverse, and scram based on comparator network using fixed setpoints
- (6) Heat exchanger -- single shell and tube simulation of the four ATR units
- (7) Secondary system pumps and throttle valve -- valve characteristics are assumed and equal flow and inertia for all pumps
- (8) Secondary throttle valve control system -- proportional and integral actions of controller are applied with temperature and flow element signal transmission losses to proper network interconnection

FIG. VII-8-1 ATR PLANT FUNCTIONAL DIAGRAM.

(9) Piping transport delays -- Pade approximation to the exact transport delay equations

(10) Primary Pressurizing and Pressure Control System -- control valve characteristics are assumed and pneumatic signal losses are simulated.

Several changes in the ATR parameters have occurred since this study was made and although these changes affect the quantitative results of this study, they are not expected to effect the general conclusions. System response tests and overall reactor transfer function experiments will supply the more exact quantitative information.

The results of the analog study show:

(1) The reactor control system can follow power demand to ± 0.5 percent over the range of 1 to 100 percent full power.

(2) The secondary throttle valve control system can control primary inlet coolant temperature $\pm 2^\circ\text{F}$ during a fast start-up and fast setback. The results also apply to 32°F cooling tower sump water temperature.

(3) The primary coolant pressure control system was able to control pressure to $+ 8.5$, -13 psig for a fast start-up and a reverse. Maximum expansion for the fast start-up was 24 gallons, and 44 gallon contraction for a reverse.

(4) Throttle valve failures, open and closed, were examined, and the control and safety systems were found capable of handling these failures.

2. SERVO REGULATOR ROD TEST PROGRAM

An ATR production unit servo regulator rod mechanism was tested to demonstrate characteristics and compatibility with reactor kinetics. The mechanism was connected to an analog computer simulation of the ATR reactor kinetics and heat transfer model.

Typical system response to a 0.1 cps sinusoidal reactivity disturbance of $0.00115 \Delta k$ amplitude (peak to peak) was

	<u>% Power Swing</u>	<u>Water Swing ($^\circ\text{F}$)</u>
Uncontrolled	7.5	4.3
Controlled	0.5	0.3

At higher frequencies (12 cps), the servo mechanism can maintain ± 0.5 percent power level control at lower amplitudes.

Step reactivity disturbances are damped within 300 ms.

A position feedback loop has been provided for manual control of the servo rod to prevent drift.

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