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SAFETY ANALYSIS REPORT

FOR THE MIT RESEARCH REACTOR (MITR-III)

(Rev. 1)

Nuclear Reactor Laboratory
Massachusetts Institute of Technology
Cambridge, Massachusetts

Chapter 1

MIT Research Reactor

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Chapter 1
MIT Research Reactor

1.1 Introduction

This Safety Analysis Report (SAR) supports an application to the U.S. Nuclear Regulatory Commission (NRC) by the Massachusetts Institute of Technology (MIT) for the relicensing of its research reactor, the MITR. In addition, a power upgrade from 5000 kW to 6000 kW is requested.

The Massachusetts Institute of Technology Reactor (MITR) is a heavy-water reflected, light-water cooled and moderated nuclear research reactor that utilizes flat, plate-type fuel elements. The MITR is owned and operated by the Massachusetts Institute of Technology, a non-profit educational institution. The MITR is the major experimental facility of the MIT Nuclear Reactor Laboratory, which is an interdepartmental laboratory that functions as a center of both education and research for many MIT departments as well as local-area universities and hospitals. The reactor facility is located on the MIT Campus at 138 Albany Street in Cambridge, Massachusetts. The MITR design includes a number of inherent (i.e., passive) safety features. The principal ones are negative reactivity temperature coefficients of both the fuel and moderator, a negative void coefficient of reactivity, the location of the core within two concentric tanks, the use of anti-siphon valves to isolate the core from the effect of breaks in the coolant piping, a core-tank design that promotes natural circulation, and the presence of a full containment. There are also several design features that are unique to the MITR. These include the use of fuel plates with finned surfaces and fuel elements with rhomboidal cross-sections. The former increases heat transfer. The latter permits the core to be in the form of a hexagonal close-packed array.

1.2 Summary and Conclusions of Principal Safety Considerations

This section summarizes safety criteria, the principal safety considerations, and the resulting conclusions of the SAR. Detailed information is given in the subsequent chapters of this report.

1.2.1 Consequences from Operation and Use

The principal conclusion of the safety analysis contained in this report is that operation and use of the MITR will result in considerable benefit without there being any significant cost to the public health or to the environment. Specific conclusions are as follows:

a) Operation and use of the MITR will provide significant benefits in terms of education, research, and the medical applications of radiation.

b) The facility's purpose is within the scope of Sections 104a (medical therapy) and 104c (research and development) of the Atomic Energy Act of 1954, as amended. Moreover, the facility will be operated in conformity with that Act, as amended.

c) Operation and use of the MITR will be conducted without consequences to the health and safety of the public. Nor will there be any significant environmental impact from the facility.

The basis for these conclusions is the careful review of the MITR design and operational safety that is documented in this SAR. Areas reviewed include the site characteristics, the design basis for facility structures, the core and the coolant systems, engineered safety features, instrument and control systems, electrical power and auxiliary support systems, and experimental facilities. In addition, administrative elements of design and operation such as radiological safety programs, operating procedures, accident scenarios, technical specifications, and financial qualifications, were reviewed as part of the process that lead to the above conclusions.

1.2.2 Safety Considerations on Choice of Site, Fuel, and Power Level

The MITR is situated on the MIT campus which is in turn located within a metropolitan area. This site was selected to promote the exchange of ideas between educators and researchers at both MIT and other local-area universities. A further reason for choosing this site was its proximity to many major medical centers. Such proximity is necessary in order to foster scientific collaboration, to provide the timely delivery of short-lived isotopes for medical research, and to permit patient therapy using neutron beams. In view of the urban setting for the MITR, a full containment structure was chosen. The building is a domed cylindrical air-tight structure. The cylindrical portion is made of 3/8-inch thick steel plate that surrounds a 2.0-foot thick concrete wall. The dome is made out of 5/8-inch steel plate. This building, which is equipped with both underpressure protection and an overpressure relief system, is designed to protect the public from radioactive effluents in the event of a facility accident. A complete description of the containment building is given in Section 6.5 of this report. The site itself is described in Chapter 2 of the SAR.

The reactor is a tank-type design that uses light water as a moderator and coolant and heavy water as a reflector. The heavy water is in a closed system. The tank-type design coupled with the separation of the moderator/coolant from the reflector offers several safety advantages. First, the reactor core is contained within a closed system. This serves to prevent the release of fission products and also provides a simple method for the detection of incipient fuel clad failures. Namely, the coolant and/or cover gas contained within the tank can be sampled for abnormal activity. Second, there are two metal tanks, that of the primary system and that of the reflector system, surrounding the core. This reduces the likelihood of a loss-of-coolant accident. Third, tritium production from the activation of heavy water is limited to the heavy-water reflector system, and even then, is not a problem because the heavy-water reflector is a closed system.

The fuel utilized for the MITR is a cermet. Such fuels provide sufficient strength for the operating environment (low temperatures and pressures) of research reactors and are resistant to the release of radioactive fission products. The fuel is made by sintering a uranium-containing metallic powder so that a certain void fraction is retained. The sintered fuel is then sealed with a clad. Fission product release is precluded by the clad. Should the clad fail, release would be slow (and hence detectable in its early stages) because the fission products would have to diffuse through the contained void spaces.

The maximum power level allowed by the Code of Federal Regulations for a research reactor is ten megawatts. Design studies have shown that the MITR is capable of operating safely at that power level provided that its heat removal equipment is upgraded. The analysis presented in Chapter 4 of the SAR demonstrates that the MITR can be safely operated with the existing heat removal equipment at a steady-state power level of 7.4 MW. The planned power level for MITR operation is 6.0 MW. This provides a 20% increase in power, and hence neutron flux, over the

MITR-II's power level of 5.0 MW. Such an increase will benefit both researchers and the facility's medical therapy capability. In addition, it maintains a substantial safety margin as established in the safety analyses. The licensed power level is 6.6 MW. This is the level at which the scrams will be set.

1.2.3 Inherent Safety Features

The MITR's design include a number of inherent or passive safety features. The principal ones are:

a) The reactor core is designed so that there are negative reactivity coefficients associated with both its fuel and moderator temperatures. These mitigate the consequences of any reactivity excursion and also promote self-regulation of the reactor. These negative coefficients arise because the core is intentionally undermoderated and its design is such that some neutron leakage occurs.

b) The reactor core is designed so that there is a negative coefficient of reactivity associated with the formation of voids. This feature also mitigates against the consequences of any reactivity excursion.

c) There should be no mixing of the light-water moderator and the heavy-water reflector because the tanks and systems are separate and independent. However, should any occur, the result will be the insertion of negative reactivity. The leakage of light water into the heavy water reflector would cause greater neutron absorption of the already-thermalized neutrons. The leakage of heavy water into the light water moderator would result in less moderation of the still fast neutrons. Hence, in both cases, the fission rate will be reduced.

d) The reactor core is contained within two concentric tanks, one for the light water that serves as both coolant and moderator and one for the heavy-water reflector. The concentric configuration of the two tanks greatly reduces the likelihood of a loss-of-coolant accident.

e) Piping penetrations to the primary core tank (the one for the coolant/moderator) are all above the core. Also, wherever appropriate, those penetrations are equipped with anti-siphon valves. As a result of both factors, a pipe break can not result in uncovering of the core.

f) The reactor is housed in a containment building that contains redundant (main and auxiliary) sets of dampers on both the inlet and exhaust ventilation penetrations. The main dampers are interlocked with the effluent radiation monitoring system and will close, thereby sealing the building, if abnormal radiation levels are detected. If these dampers fail to close within a specified time interval, the auxiliary dampers are signaled to close. The main dampers are hydraulically operated. The auxiliary dampers are gravity-operated. Both sets close on loss of electricity.

g) The geometrical arrangement of the core relative to the core housing and tank was chosen to promote natural circulation in the event of a loss-of-flow accident. In addition, there are specially designed natural circulation valves that serve to enhance the establishment of heat removal in the absence of forced circulation of the coolant.

h) The reactor control blades, which are withdrawn to attain criticality, are held in place by electromagnets. Upon de-energizing the magnets, the blades drop by gravity into the core thereby making it subcritical.

i) The reactivity worth of the heavy water reflector is substantial and its removal will make the reactor subcritical even if the control blades are still withdrawn. The reflector can be rapidly "dumped" to a holding tank by means of a solenoid-controlled air-operated valve. Loss of electricity, loss of air, or manual action will result in a reflector dump.

j) The containment building is protected against vacuum forces by two sets of redundant spring-loaded vacuum relief breakers.

1.2.4 Design Features for Safe Operation and Shutdown

There are a number of design features and design bases of systems and components associated with the MITR that promote safe operation and shutdown. These include:

a) Reactor Control System: The reactor control system consists of both shim blades and a regulating rod. The former are used to attain criticality, to compensate for fission product poisoning and fuel depletion, and to accomplish major changes of reactor power. The latter is used for the fine adjustment of reactor power. The six shim blades are connected to drive mechanisms by electromagnets. Hence, on loss of electricity, all blades drop into the core under the influence of gravity. The reactor control system restricts reactivity insertion by permitting only one shim blade to be withdrawn at a time. All six blades can be driven in simultaneously to shut the reactor down. In an emergency, all six can be simultaneously dropped into the core.

b) Reflector Dump Capability: The heavy-water reflector can be dumped to a holding tank. Loss of the reflector shuts the reactor down, so this capability is a backup shutdown mechanism.

c) Reactor Protection System: The reactor protection system consists of the nuclear and process safety systems. It monitors parameters that are important to safety including the reactor power and period, coolant outlet temperature and flow, and the core tank level. The system uses a one-out-of-one logic. Hence, even if a particular signal is measured by redundant instruments, any single out-of-specification signal will cause a shutdown. Actuation of the protection system causes the shim blade electromagnets to be de-energized, thereby dropping the six shim blades into the core.

d) Emergency Core Cooling System: Passive safety features (the natural convection and anti-siphon valves) are sufficient to ensure that the core is adequately cooled upon a loss of forced convection. Also, the presence of both a core tank and a reflector tank as well as the use of anti-siphon vales greatly reduces the likelihood that the core would ever be uncovered. Nevertheless, an emergency core cooling system has been installed. This system, which requires manual initiation, sprays water on to the core at a rate sufficient to remove decay heat.

e) Containment Building Pressure Differential: The containment ventilation system consists of intake and exhaust fans and a control damper. The control damper is adjusted to maintain building pressure slightly below atmospheric pressure. Hence, any air leakage will be into the building rather than out of the building.

f) Containment Building Isolation: Operation of the ventilation exhaust dampers, which close on loss of electricity, was described above in Section 1.2.3(f) of this report. An engineered safeguards feature is associated with this system. The exhaust ventilation flows through a holdup plenum. Effluent radiation monitors are located at the entrance to this plenum and the ventilation system's isolation dampers are located at the exit. The air flow delay through the plenum is such that if abnormal activity is detected, the building will be sealed by the isolation damper system before it can be released.

g) Containment Pressure Relief: The containment building is equipped with a pressure relief system that processes exhaust air for the removal of radioactive particulates and iodine. This system can be placed on line manually without entry to the containment building. Once on line, flow through the system is the result of the pressure difference between the building and the environment. Its purpose is to permit building pressure to be maintained at or below a pre-determined allowed value in the event that the building is sealed.

1.2.5 Potential Accidents

The maximum hypothetical accident (MHA) for the MITR is that some foreign material becomes lodged beneath the core where it causes partial blockage of fuel element channels. Because of the size of the openings in the nozzles at the end of each fuel element, any piece of foreign material that could pass through the nozzles would not be large enough to restrict flow in more than five coolant channels of one element. Hence there could be, at most, the overheating of a maximum of four fuel plates. The MHA conservatively assumes that the entire active portions of all four plates melt completely. Analysis of this accident is given in Section 13.2.1 of this report. The calculated maximum external dose to an individual located at the nearest points of public occupancy during the first two hours following the MHA (the time allowed for protective actions to be implemented) is 197 mrad at 8 m (back fence) and 247 mrad at 21 m (front fence) to the whole body. The maximum whole body dose is 300 mrem at 16 m. The internal dose is 135 mrad and 134 mrad to the thyroid at 8 m and 21 m, respectively.

Other accidents that are evaluated in Chapter 13 of this SAR include insertion of excess reactivity, loss of primary coolant, loss of primary coolant flow, malfunction of fuel, loss of normal electrical power, external events, and equipment malfunctions. These evaluations demonstrate that none of these accidents will result in a safety hazard to the public or the environment. It should be recognized that some of these accidents, such as a loss of coolant flow or a loss of offsite electricity, have occurred and are expected to occur again. The MITR was designed with this expectation so that no safety hazards or environmental effects will result.

1.3 General Description of the Facility

The original MIT Reactor, MITR-I, was both heavy-water moderated and cooled with an open array of plate-type fuel elements. It was surrounded by a graphite reflector. This original core attained criticality in July 1958 and operated at power levels of up to 5 MW until May 1974. The present core differs significantly from the original design in that it uses light water to cool and moderate a close-packed array of finned, plate-type elements that are surrounded by a heavy-water reflector which is in turn surrounded by the original graphite reflector. This core design was chosen to maximize the thermal neutron flux in the reflector regions where the experimental beam ports are located.

The present MITR core has several unique features. First, the fuel plates are finned to increase heat transfer. Second, the fuel elements are rhomboid in cross-section. This allows construction of a hexagonal shaped core that contains twenty-seven elements grouped in three rings. These are an A-ring of three elements, a B-ring of nine elements, and a C-ring of fifteen elements. Third, the control devices (six shim blades and a fine-control regulating rod) are all located on the core periphery in the space between the C-ring and the light-water core tank.

Fourth, beamports are aligned so that they are opposite a position below the core where the thermal flux is at a maximum because of water peaking effects. Those locations also improve the quality of the flux provided to experimenters because all neutrons entering a beam port must have undergone at least one thermalizing collision.

The MITR is located in the northwest section of the MIT campus in Cambridge, Massachusetts. The site is urban with activities related to both manufacturing and retail sales located nearby. In addition, there are other MIT-owned properties adjacent to the reactor site. These include administrative and teaching facilities, dormitories, and other laboratories. An advantage to this urban setting is the proximity of the site to security, fire, and medical assistance should those services be needed.

The meteorological and seismic characteristics of the site are those of the northeastern United States. Hurricanes and winter storms do occur. However, extreme weather conditions, such as tornadoes, are not a factor. There has been low to moderate seismic activity in historic time both to the north and south of the Cambridge-Boston area. However, Cambridge itself lies in a basin that has been largely free of such activity. The MITR site is drained by the Charles River immediately before it flows into the Atlantic Ocean. Hence the site is not subject to flooding.

The principal design criteria is that reactor operation should not result in any fuel damage. As a practical matter, this means that the thermal-hydraulic, reactor control, and reactor protection systems are designed to prevent the occurrence of "critical heat flux" or CHF. To assure further the safe operation of the MITR, limiting safety system settings (LSSS) are selected to prevent the onset of nucleate boiling anywhere in the core. For the transients listed above in Section 1.2.5 of this report, a reactor scram at the LSSS will prevent the occurrence of CHF.

The MITR is capable of operating at a steady-state power level of 7.4 MW (7400 kW) It is planned to operate at 6.0 MW (6000 kW) thereby providing a substantial safety margin. The reactor is not designed for pulsed operation. Its principal operating characteristics are negative coolant and fuel temperature coefficients of reactivity, a negative void coefficient, a negative change of reactivity should there be any accidental mixing of the light-water moderator and the heavy-water reflector, the inability to attain criticality on withdrawal of a single control device, and the capability to make the facility subcritical by an amount equal to or greater than the required shutdown margin through use of either the control devices or the option to dump the heavy-water reflector.

The principal safety systems are (1) the reactor protection system, which is used to monitor both core neutronic performance and the status of the various process systems especially those associated with heat removal; (2) the reactor control and reflector dump system, which provides the means to make the reactor subcritical; and (3) the containment ventilation isolation and pressure relief systems, which ensure the retention of any abnormal radioactivity within the containment building itself.

The reactor is equipped with a number of passive and engineered safety systems. The former were enumerated above in Section 1.2.3 of this report. The latter include:

a) Emergency core cooling that, upon manual initiation, sprays water onto the core to remove decay heat.

b) Emergency electrical power that, upon loss of the normal electrical power supply, picks up loads important to safety.

c) Containment isolation that, upon actuation by an effluent radiation monitor, seals the building.

d) Containment pressure relief that, upon manual initiation, relieves pressure in the containment building by discharging air through filters that remove particulates and iodines.

The MITR has five major instrumentation systems. These are the reactor control system, the reactor protection system, engineered safeguards actuation system, the control console display instruments, and the radiation monitoring system. These systems utilize redundancy to avoid single-point failures and diversity to preclude common mode failures. Most instruments indicate remotely in the reactor control room. Signals that are of importance to the operator-in-charge for the manipulation of reactor power are displayed on the console. These include indications of reactor power and period, control device position, coolant temperature, and process system flows. Also provided at or near the console are readouts from the effluent and area radiation monitors, an annunciator panel that reflects the status of the process systems, and the outputs of the nuclear safety system.

Control of the MITR is at all times under the auspices of a licensed reactor operator. That operator may, depending on the circumstances, either perform the needed manipulations manually or employ an automatic controller. The principal modes of control are:

a) Startup: Upon completion of mechanical and instrumentation checklists, the operator may take the reactor critical and raise its power level. This is done by withdrawing each of the shim blades, one at a time. The withdrawals are done in small increments with each blade being moved in sequence so as to maintain a uniform bank height. A number of interlocks must be satisfied in order to initiate a startup. These ensure that the requisite initial conditions (e.g., all process and instrumentation system operating) are satisfied.

b) Power Adjustment: Major adjustments of reactor power are normally performed manually by moving one of the shim blades so as to establish the requisite period. Movement of the blade is limited to a certain increment (either above or below) the bank height. Once the new power level is attained, the blade is returned to the bank and the regulating rod adjusted so as to make the reactor exactly critical.

c) Steady-State: Steady-state operation is usually achieved via an automatic controller that moves the fine-control regulating rod in response to a deviation between the measured power and a setpoint. Small power adjustments can be made on automatic control by slowly changing the setpoint. In the event of a sudden large deviation, control will trip automatically to manual and sound an alarm. Steady-state operation can also be done manually.

d) Shutdown: Reactor shutdown can be accomplished in several different ways. First, the operator can perform a shutdown manually by driving each blade in sequentially in small increments so as to maintain a uniform bank height during the shutdown. This is normally not done. Second, all blades can be inserted simultaneously at their normal speed. This is the preferred mode. Third, the reactor can be scrammed. Two types of scrams are available. A "minor" scram causes all blades to drop. A "major" scram does that and also both dumps the reflector and seals the containment building. Fourth, the reflector can be dumped without other action such as building isolation.

e) Digital Automatic Control: The MITR has approval for the digital control of both its regulating rod and its shim blades. Two types of controllers are possible. For the first category, safety is assured by limiting the net reactivity worth of the control element (blade or regulating rod) that is associated with the automatic controller. For the second category, safety is assured by designing the controller in accordance with the concept of "feasibility of control." (See Section 10.2.10 of this report.) This ensures that it will always be possible to achieve some equilibrium state before a limiting condition is attained.

Offsite electricity is supplied by two 13.8 kV power lines that feed separate 1200 ampere circuit breakers. These breakers are interlocked so that only one is closed at a time. A transformer reduces the voltage to 480 V and supplies separate load centers. One load center provides electrical power to the Nuclear Reactor Laboratory's administrative building. The others supply various reactor loads. In addition to the normal electrical power supply, there is an emergency distribution system. A storage battery supplies a motor-generator (MG) set that in turn provides AC power to vital loads. On loss of normal off-site power, the MG set starts automatically and essential loads transfer to it. These include the minimum complement of instrumentation needed to ensure reactor safety, the minimum complement of radiation monitors, essential lighting, communications equipment, and the decay heat removal pump.

Heat produced from fission in the MITR core is deposited in the primary, heavy-water reflector, and shield systems. All three are equipped with heat removal systems. The primary heat removal system is the largest because it transports the bulk of the fission energy. It consists of two centrifugal pumps, heat exchangers, a cleanup loop, and a storage tank that both allows for volume changes and provides a reserve. In addition to removing heat from the core, primary coolant is circulated through the light-water shutter that controls admission of the neutron beam to the medical irradiation room that is located in the reactor basement. Primary coolant from the storage tank can also be supplied to the emergency core cooling system although the normal lineup for this system is from city water.

The coolant systems for the heavy water reflector and shield system are similar except that they each utilize a single pump and heat exchanger. Each has its own cleanup loop and storage tank.

The MITR generates high-level waste (spent fuel) and low-level waste. The latter may be solid, liquid, or gaseous. The spent fuel is returned to the U.S. Department of Energy as discussed below in Section 1.7 of this report. The low level waste is treated as follows:

a) Solid waste (rags, dewatered resins, contaminated tools, etc.) is compacted, and either stored on-site for decay or shipped off-site to an approved disposal site.

b) Liquid waste is held in one of two storage tanks for decay. It is then recirculated to ensure uniformity, filtered, sampled, and discharged to the sanitary sewer.

c) Gaseous waste is filtered for the removal of particulates and iodines and then discharged via the containment building stack.

Both liquid and gaseous discharges are continuously monitored for radiation level. Interlocks will halt the discharge if any abnormal activity is detected.

A radiation protection program for those who work at the facility is designed and administered by an on-site group that is part of the MIT Radiation Protection Office. That Office reports to the MIT Medical Department and hence is independent of Reactor Operations.

The MITR is equipped with a number of experimental facilities. The principal ones are:

a) Medical Irradiation Rooms: Two such facilities are available. One is located on the basement and uses a neutron beam from the reactor core. The other is located on the reactor floor and uses a beam produced by a fission converter that in turn is driven by neutrons from the core. The basement facility's beam may be either thermal or epithermal depending on the filters that are used. The fission converter beam is epithermal. Both facilities are used to support research in the medical uses of neutrons in cancer therapy.

b) Beam Ports: Numerous beam ports penetrate the reactor's shield, graphite reflector, and heavy-water reflector. These provide a high-quality neutron flux for such endeavors as neutron scattering, prompt-gamma analysis, and neutron physics.

c) Automatic Sample Transfer: The MITR is equipped with pneumatically-operated horizontal tubes that allow samples to be exposed to various neutron fluxes. These facilities are normally used to support the neutron activation analysis of environmental samples such as air pollutants.

d) Graphite Reflector: The MITR is surrounded by a graphite reflector. Vertical irradiation thimbles penetrate this reflector. These are normally used for the activation of geological specimens that require a uniform, thermal flux.

e) In-Core Sample Assemblies: There are twenty-seven fuel element positions. Of these, only twenty-three or twenty-four, depending on fuel burnup, are required to achieve criticality. The other three or four positions can be occupied by in-core sample assemblies. These are normally used for major experiments such as in-core loops that replicate conditions in pressurized or boiling water reactors.

1.4 Shared Facilities and Equipment

The MITR is located within its own containment building. As a result, it shares few systems or equipment with other facilities. Those that are shared are as follows:

a) Electricity Supply: Off-site electricity for both the reactor building and an adjoining administrative building is brought into the utility area that adjoins the containment building. The electric power is distributed among six circuits. Five of these supply the reactor and its associated load centers. The sixth provides electrical power to the administrative building.

b) Water Supply: Off-site water is supplied to the reactor building and cooling tower makeup system. A backflow preventor precludes the possibility of any water being returned to the potable water supply.

c) Heating: Heat for the reactor containment building is provided from a source that also supplies other MIT buildings located in the northwest section of the campus.

d) Compressed Air: The reactor building has a dedicated supply of compressed air. Should that supply be lost, the distribution network can be cross-connected to a supply that is normally used for nearby laboratories.

e) Sanitary Sewer: Liquid discharges from the reactor building are directed, after processing, to a common sanitary sewer.

In addition to the shared items, certain other services originate outside the containment building even though they are dedicated systems. These include the helium and CO_2 cover gas supplies, purified water for use as makeup, and a capability to transfer irradiated samples via pneumatic tubes. Each of these systems is equipped with either manual or solenoid-operated valves, or both, so that containment integrity is maintained.

There is a hot cell facility located within the containment building on the main floor. It is used to service in-core experiments. It has its own ventilation, fire-protection, and radiation-monitoring system.

1.5 Comparison with Similar Facilities

The MITR is one of twenty-seven operating research reactors that are located on university campuses. Others are located at hospitals, industrial sites, and research laboratories. In addition, there are more than fifty research reactors operating overseas. The power levels of these facilities range from a few watts to 10 MW, the maximum allowed value for a research reactor. In general, research reactors may be divided into two groups: those that use TRIGA fuel and those that use MTR fuel. The former often have a pulse capability whereas the latter do not. The MITR is among the latter group. The safety record for both types of research reactor has been excellent. This reflects both the actions of the individual licensees and the robust design of these reactors.

1.6 Summary of Operation

The mission of the MITR is to support the educational and research programs of the Massachusetts Institute of Technology. A related objective is to support similar programs at other local-area universities and hospitals. To this end, the MITR currently operates twenty-four hours per day. A normal cycle is for the reactor to run at full power for four weeks and then to shut down for several days in order to conduct tests and calibrations, to perform maintenance and refuelings, or to remove or install experiments. The major experimental capabilities include:

a) In-core loops that replicate the environment of pressurized or boiling water reactors. The objectives include the study of mechanisms for materials failures and methods for improving water chemistries.

b) Medical therapy facilities for the treatment of certain cancers such as metastic melanoma and glioblastoma multiforme using neutron capture therapy.

c) Beam tubes for the conduct of neutron scattering experiments.

d) Irradiation facilities for the conduct of neutron activation analysis and for the production of short-lived medical isotopes.

1.7 Nuclear Waste Policy Act of 1982

The MIT Research Reactor (MITR) is owned and operated by the Massachusetts Institute of Technology, a non-profit university. Fuel for the operation of the MITR (and also the heavy water for the reflector) was originally provided by the U.S. Atomic Energy Commission (AEC). The U.S. Department of Energy (DOE) is the successor agency to the AEC and it now provides both the fuel and heavy water. It also retains title to both. It is MIT's understanding that DOE (R.L. Morgan) has informed NRC (H. Denton) of the title arrangement for the fuel in a letter dated May 3, 1983. Specifically, MIT and DOE have a contractual arrangement whereby DOE retains title to the fuel and is obligated to take the spent fuel and/or high-level waste for storage or reprocessing. All of the MITR-I fuel and much of the MITR-II spent fuel has been returned to DOE pursuant to this arrangement. A copy of the contract between MIT and DOE for fuel assistance is available from MIT's Office of Sponsored Programs.

1.8 Facility Modifications and History

On May 7, 1956, the United States Atomic Energy Commission (AEC) issued Construction Permit No. CPRR-5 to the Massachusetts Institute of Technology (MIT). Construction then began on the original reactor, the MITR-I. The MITR-I was heavy-water cooled and moderated. On June 9, 1958, the AEC issued Facility Operating License No. R-37, which authorized operation at power levels up to 1 MW. The license was effective on its date of issuance and it was to expire at midnight May 7, 1996. Initial criticality was achieved on July 21, 1958. In June 1961, an increase in the allowed operating power to 2 MW was approved. In October 1965, a further increase to 5 MW was authorized. On April 9, 1973, Construction Permit No. CPRR-118 was issued to MIT by the AEC. This permit authorized the modifications to create the present MITR. At 1618 May 24, 1974, the original MITR-I was shut down for the last time and further operation was precluded until the construction specified in Permit No. CPRR-118 was complete. Permit No. CPRR-118 authorized modification to a light-water cooled and moderated facility with a heavy-water reflector. The new design, known as the MITR-II, offered higher flux levels for the same power as well as significantly reduced tritium production. On July 23, 1975, the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 10 to Facility Operating License No. R-37. This amendment authorized operation of the modified reactor at power levels up to 5

MW. Initial criticality of the MITR-II was achieved on August 14, 1975. Significant regulatory actions that occurred after 1976 were the approval to conduct digital control experiments on the MITR, the authorization to use the medical facility for neutron capture therapy for humans, the safety evaluation of a fission converter facility, and the extension of the operating license to August 1999 to recover time spent in construction.

Chapter 2

Site Characteristics

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2.1 Geography and Demography

The geography and demographics of the location selected for the MITR site are described here.

2.1.1 Site Location and Description

The MITR was sited on the MIT campus in order to permit daily use of the facility by MIT faculty and students as well as researchers from other universities and local-area hospitals. Such proximity is essential to the exchange of ideas, to progress in research activities, and to the education of students. Access to experimental facilities is of particular importance in teaching because it is through the experience of working as a team on a research project that students acquire the skills to conduct their own research or to direct industry research programs. A further advantage to this site is that it makes the facility accessible to other Boston-area universities and to high schools located throughout eastern Massachusetts. Many of these schools regularly tour the facility.

The conclusion reached in both this chapter and in the other chapters that comprise this report is that the site is well-suited for the MITR given the benign characteristics of the reactor. In particular, it operates at low power (a factor of 300 less than most electric power-production facilities), at atmospheric pressure, and at low temperature (hot bath tub water). Hence, there is neither a large inventory of radioactive fission products nor the stored thermal energy to disperse that inventory. Also, the facility has a full containment.

The MITR is, as of 1998, one of 27 operating research reactors that are located on university campuses. Additional research reactors are located at hospitals and industrial sites. Also, there are more than 50 research reactors operating overseas.

2.1.1.1 Specification and Location

The MIT Research Reactor, MITR, is located at latitude 42° 21' 28" and longitude -71° 5' 47". The corresponding Universal Transverse Mercator (UTM) coordinates are Zone Number 19, Northing 4691629 m, and Easting 327354 m. This location is on the northwest side of the MIT campus which is located in Cambridge, Middlesex County, Massachusetts. The most prominent natural or man-made feature in the immediate vicinity is the Charles River Basin which forms the southern border of the MIT campus.

2.1.1.2 Boundary and Zone Area Maps

Figure 2-1 is a map of Massachusetts on which the location of the MITR is shown together with that of major cities. Also shown are the two other university research reactors that are located in the state. These are the Worcester Polytechnic Institute and the University of Massachusetts at Lowell. Figures 2-2 and 2-3 show the reactor location with respect to the Cambridge-Boston metropolitan area. Figure 2-4 is a map of the MIT campus. This shows the general location of the MITR. Other major laboratories that are located on or near the campus include the MIT Plasma Fusion Center (PFC), the Whitehead Institute, and the Charles Stark Draper Laboratory. Figure 2-5 is an enlargement of the area occupied by the MITR in Figure 2-4. It shows the site boundary.

The reactor site, as shown in Figure 2-5, measures approximately 86 m (280 ft.) in length and 46 m (150 ft.) in width. Included in this area are the reactor building, an adjoining one-story building that houses laboratories and storage areas as well as reactor support equipment such as electrical switchgear, Building NW12, the cooling towers, the ventilation exhaust stack, and a parking lot. Building NW12 contains offices, classrooms, laboratories, and auxiliary equipment associated with the reactor. The reactor building is connected to the one-story building by an air lock. Entry to structures and areas inside the one-story building, the reactor containment building, and the area occupied by the cooling towers and ventilation exhaust stack is restricted.

The reactor site is in the midst of a heavily commercialized section of Cambridge. CONRAIL railroad tracks, used exclusively for freight traffic, run parallel to the back of the reactor site. Across the tracks, at a distance of about 25 m (80 ft.), is a storage warehouse that extends the length of the site. MIT Building NW13, which is presently used for nuclear engineering projects for nuclear chemistry studies, is about 46 m (150 ft.) from the reactor building and is immediately adjacent to the side of the site away from Massachusetts Avenue. On the Massachusetts Avenue side of the site, about 25 m (80 ft.) away from the reactor building, is an MIT-owned parking lot. At the front of the site, approximately 21 m (68 ft.) from the reactor building is Albany Street. Although the site boundary comes nearest to the reactor on the side facing the railroad tracks, the closest point of normal public occupancy near the site boundary is on the Albany Street side. Across Albany Street is a parking lot and facilities of a local commercial concern. Also across Albany Street but to the left of the reactor site is an MIT dormitory for graduate students. The area behind both the dormitory and the commercial concern is University Park which includes retail shopping and a hotel.

The following boundaries are important from the standpoint of normal and emergency operation.

a) <u>Operations Boundary</u>: This is the area within the site boundary where the reactor's chief administrator has distinct authority over all activities. It includes the reactor containment building, the adjoining one-story building, and the area occupied by the cooling towers and ventilation exhaust stack. The latter area is enclosed by a perimeter fence. Access to this area is restricted to those so authorized by management of the MITR.

b) <u>Site Boundary</u>: The site boundary is that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee, wherein the reactor administrator may directly initiate emergency activities. The area within the site boundary may be frequented by people unacquainted with the reactor operations. The site boundary of the MIT Research Reactor encompasses the restricted area, (the area within the operations boundary), the reactor parking lot, and all of Building NW12. (<u>Note</u>: The parking lot is between the containment building and Albany Street. Parking is limited to authorized vehicles only.)

c) <u>Emergency Planning Zone</u>: The Emergency Planning Zone or EPZ is a 100 m radius surrounding the reactor containment building. This zone includes MIT-owned buildings, other privately held property, and public streets.

d) Nearby Structures: The reactor's 150 ft. ventilation exhaust stack is the tallest structure in the local area. The nearest tall structure in the vicinity of the MITR is MIT's Green Building which is about a half-mile away in the central part of the campus. It is 295 ft. tall and houses the Department of Earth and Planetary Sciences which can provide weather-related information to the rest of MIT. The next nearest structures that are taller than the stack are located in East Cambridge at a distance of approximately one mile.

In terms of topography, Cambridge is located in a basin that is formed by a low-lying chain of hills. The basin is drained by the Charles River which flows into Boston Harbor and Massachusetts Bay. Drainage is therefore eastward toward the ocean. Figure 2-3 shows the reactor site in relation to the Cambridge-Boston area.

2.1.2 Population Distribution

The City of Cambridge has a population of 95,802 according to the 1990 census. The population of Boston in 1990 was 574,283. The Cambridge population has remained relatively constant over the past twenty years, with the Boston population showing a decrease as the population has shifted towards the suburbs. The greater Boston population in 1990 was 2,870,650, which is slightly higher than the 1965 population of 2,605,452 [2-1].

Cambridge has a high population density of 14,891 persons/square mile. Figure 2-2 shows a map of the greater Boston area with concentric circles at 1, 2, 4, 6 and 8 kilometers from the reactor.

The area enclosed within the 1 km circle around the reactor includes all of MIT, some factories and commercial facilities, and a residential section. There are normally 4150 students living on the MIT Campus. Approximately two-thirds of these students are undergraduates who reside on campus only during the school year, which is September through May. The residential section, mostly north of the MIT campus, comprises about one-third of the 1 km circle. If a uniform population density distribution is assumed throughout the Cambridge residential areas, the residential population is about 6090 persons [2-2]. This is a conservative estimate because about one-third of the area is zoned for commercial use. Combined with all of the MIT on-campus residents, the total number of residents in the 1 km area is approximately 10,300. The Metropolitan Area Planning Council forecasts that the Cambridge population will remain relatively constant through the year 2020 giving an overall change of less than 0.2% [2-3].

The 2 km radius includes about one-third of the area of Cambridge plus some residential areas of Boston along the Charles River as well as a small residential area of Somerville. If the population density throughout all of this area is assumed equal to the average population density of Cambridge (again this is conservative with both the industrial sections and the Charles River included), the population in this area is about 73,000 persons. The population in this area should also remain relatively constant throughout the reactor license period.

The area enclosed by the 4 km radius includes parts of the following cities: Cambridge, Somerville, Brookline, and the Charlestown, Beacon Hill, Back Bay, South Bay, Allston, and Lower Roxbury sections of Boston. Taking the relative percentages of the area of these cities and assuming a constant population density for each city as given in the 1990 census, gives a total population estimate of this area of about 264,000. The Metropolitan Area Planning Council forecasts about 5% population increase in Boston through the year 2020. Because Cambridge and some of the other adjacent cities are predicted to have a flat or negative population growth, the total population change for the 4 km radius will probably be somewhat less than 5% through the end of the licensing period.

At 6 km from the reactor, all or part of the following cities are included: Cambridge, Somerville, Medford, Everett, Chelsea, Watertown, Belmont, Arlington, Brookline, and about one-half of Boston. The total population estimate of this area is about 570,000, which should increase by about 5% through the year 2020.

The 8 km zone encloses all of Cambridge, Everett, Somerville, and Chelsea as well as parts of ten other cities, including most of Boston. The total population estimate of this area is about 850,000, again estimated to increase by 5% through 2020.

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 Locations and Routes

A CONRAIL branch runs directly behind the reactor site. These tracks are currently used only for freight traffic. Trains are infrequent, usually two or three per day.

Motor vehicle traffic is generally heavy at all times. Massachusetts Avenue, which parallels the site on the east, is a major route for all types of vehicles. Memorial Drive, which runs along the Charles River, and hence to the south of the reactor site, is restricted to passenger vehicles.

The Boston Inner Harbor is about 5 km from the MITR and ocean-going vessels that enter the harbor usually dock at Charlestown or East Boston. Cargoes do include LNG and oil.

Logan International Airport is about 6 km to the east of the MITR. Air traffic is not normally routed over Cambridge. This is further discussed below in Section 2.2.2 of this report.

2.2.2 Air Traffic

The General Edward Lawrence Logan International Airport is located about 6 km east of the MIT Reactor, in East Boston, which is across the Boston Inner Harbor from the

Charlestown and downtown areas of Boston. Logan Airport serves greater Boston and handles about 510,000 aircraft movements per year, ninety-five percent of which are commercial.

The airport contains ten major runways, with eight of the approach patterns being generally either from the north or the south. Only one of the runways, runway 09, is approached from an easterly direction, with the approach pattern being over Boston, not Cambridge. The closest airport approach pattern to the reactor site for any type of aircraft is a helicopter approach pattern which runs along the northern edge of Boston, which is over 1 km from the reactor site at the closest point. Aircraft holding patterns are normally well away from both the airport and the Cambridge area [2-4]. Thus it is very unlikely that an accident involving an aircraft following normal flight patterns would affect the reactor. (Note: Logan Airport is currently seeking permission for construction of a second runway in the easterly direction.)

2.2.3 Analysis of Potential Accidents at Facilities

To summarize the discussion given in Section 2.2.1 and 2.2.2 of this report, the MITR is in an urban setting and certain hazards associated with the use and transport of toxic materials exist. These hazards are regulated by several agencies, primarily by U.S. Department of Transportation, to ensure safety. The management of the MITR meets regularly with the Director of Emergency Planning for the City of Cambridge. One objective of these meetings is to keep the MITR staff informed of any potential hazards to the reactor.

2.3 Meteorology

2.3.1 General and Local Climate

The dominant characteristic of the weather in the metropolitan Boston area is that it changes frequently. The region is in a transition zone of prevailing west to east atmospheric flow. Most low-pressure systems that either enter or are created within the United States exit through the Northeast and hence affect the Boston weather. The dominant air masses are: (1) the polar continental, which are generally cold and dry and originate in the Canadian and Arctic areas: (2) the maritime tropical, which are warm and moist and originate in the Gulf of Mexico and adjacent tropical waters; and (3) the polar maritime, which are cool and damp and originate in the northern Atlantic Ocean. Movement of the polar continental and polar maritime masses to the south and movement of the maritime tropical masses to the north adds to the variability of the weather. There is no dry season. Summer precipitation is in the form of showers and thunderstorms. Winter precipitation is often as snow [2-5].

While the weather is subject to frequent variability, extremes are unlikely because the proximity of the Atlantic Ocean mitigates seasonal temperature variation. Hurricanes do occur in late August and September. However, tornadoes are virtually non-existent.

Wind rose data for the Boston area is given in Figure 2-6 and Table 2-1 for 1989. Also shown (Table 2-2) is an average for five years. This information is provided by the Environmental Protection Agency. The wind blows most frequently from a westerly direction, with southwest winds predominating in the summer and fall months and northwest winds predominating in the winter and spring months. The yearly mean wind speed is 20 km/hr (12.5 mph), being slightly higher in the winter and lower in the summer.

Temperatures are somewhat moderated by the Atlantic Ocean and vary from a normal January low of 21.6°F (-5.8°C) to a normal July high of 81.8°F (27.7°C). Extremes have varied from -12°F (-24°C) in January of 1957 to 102°F (39°C) in July of 1977. The temperature normals and extremes are shown in Table 2-3.

Boston has no dry seasons, with relative humidity remaining somewhat constant throughout the year at 60% to 70%, being slightly higher in the fall. The greatest amount of precipitation is caused by frontal activity and involves a tropical maritime air mass containing moisture being pushed up over a cold front, lifting the warm air above condensation levels. The wettest months are most frequently November or December with an average monthly water equivalent precipitation of slightly over four inches. Summer months are somewhat drier, with an

Table 2-1

Boston Logan Airport Wind Distribution - 1989

Station ID: 14739
Year: 1989
Start Date: January 1
Start Time: Midnight

Run ID: Boston, MA 1989

End Date: December 31
End Time: 11 PM

Frequency Distribution

Speed (knots)

	1-3	4-6	7-10	11-16	17-21	≥21	Total
N	.000571	.008219	.018950	.017808	.004795	.000342	.050685
NNE	.000457	.006963	.011530	.007420	.002626	.000114	.029110
NE	.000228	.005822	.012100	.011530	.004338	.002397	.036416
ENE	.000571	.005822	.015868	.018607	.005479	.000457	.046804
E	.000342	.005822	.020434	.020434	.002169	.000228	.049429
ESE	.000228	.007078	.022374	.021119	.002626	.000228	.053653
SE	.000228	.007192	.022945	.010160	.001027	.000228	.041781
SSE	.000685	.007763	.014384	.005365	.000114	.000228	.028539
S	.000685	.008333	.022032	.013584	.002055	.001370	.048059
SSW	.000228	.008447	.024772	.028653	.007192	.001484	.070776
SW	.000114	.009018	.035731	.042922	.011986	.002397	.102169
WSW	.000228	.004680	.028082	.037215	.007763	.001370	.079338
W	.000000	.003539	.014726	.032877	.009475	.001027	.061644
WNW	.000228	.003311	.032534	.065525	.019863	.005365	.126826
NW	.000228	.004795	.030137	.054224	.015183	.003539	.108105
NNW	.000114	.005023	.019863	.030365	.007306	.000457	.063128
Total	.005137	.101826	.346461	.417808	.103995	.021233	

Frequency - winds calm .003539

Source: U.S. Environmental Protection Agency [2-6].

Table 2-2

Boston Logan Airport Wind Distribution - Five Year Average

Station ID: 14739
Years: 1985 - 1989
Start Date: January 1
Start Time: Midnight

Run ID: Boston, MA

End Date: December 31
End Time: 11 PM

Frequency Distribution

	Speed (knots)						
	1-3	**4-6**	**7-10**	**11-16**	**17-21**	**≥21**	**Total**
N	.000639	.009470	.024621	.017068	.003012	.000388	.055198
NNE	.000730	.007827	.011364	.008534	.001666	.000183	.030303
NE	.000570	.006709	.012003	.014376	.005545	.001825	.041028
ENE	.000685	.006549	.015243	.014558	.003172	.000548	.040754
E	.000456	.006480	.018802	.016087	.002213	.000890	.044930
ESE	.000525	.006823	.020172	.019213	.002373	.000365	.049471
SE	.000525	.008009	.022180	.010268	.000821	.000297	.042100
SSE	.000479	.007142	.012505	.003674	.000342	.000183	.024325
S	.000799	.009150	.020400	.011957	.001460	.000730	.044496
SSW	.000479	.008397	.023800	.026766	.005020	.001050	.065512
SW	.000411	.008922	.038221	.044907	.007918	.001255	.101634
WSW	.000753	.006777	.032357	.041461	.005887	.000662	.087897
W	.000411	.004906	.023435	.038267	.007530	.001392	.075940
WNW	.000411	.006229	.038335	.059488	.014558	.003081	.122102
NW	.000479	.006549	.033543	.049242	.013531	.002419	.105764
NNW	.000251	.005568	.024621	.025945	.005340	.000753	.062477
Total	.008603	.115507	.371600	.401812	.080390	.016019	

Frequency - winds calm .006070

Source: U.S. Environmental Protection Agency [2-6].

Table 2-3

Temperature Normals and Extremes

Temperature °F	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sept.	Oct.	Nov.	Dec.	Year
Normals													
– Daily Maximum	35.7	37.5	45.8	55.9	66.6	76.3	81.8	79.8	72.8	62.7	52.2	40.4	59.0
– Daily Minimum	21.6	23.0	31.3	40.2	49.8	59.1	65.1	64.0	56.8	46.9	38.3	26.7	43.6
– Monthly	28.6	30.3	38.6	48.1	58.2	67.7	73.5	71.9	64.8	54.8	45.3	33.6	51.3
Extremes													
– Record Highest	63	70	81	94	95	100	102	102	100	90	79	73	102
– Year	1990	1985	1985	1976	1979	1952	1977	1975	1953	1963	1994	1984	Jul 1977
– Record Lowest	-12	-4	6	16	34	45	50	47	38	28	15	-7	-12
– Year	1957	1961	1984	1982	1956	1986	1988	1986	1965	1976	1989	1980	Jan. 1957

Source: National Climate Data Center [2-7].

average monthly precipitation of about three inches. The largest monthly rainfall on record was 43.4 cm (17.1 inches) in August of 1955 and the least was 0.9 cm (0.35 inches) in September of 1957. Snowfall typically occurs from November through April, with February usually seeing the greatest amount. The largest monthly snowfall on record is 105 cm (41.3 inches) in February of 1969. The largest snowfall in 24 hours was 60 cm (23.6 inches) in February of 1978.

The weight of the 100 year return snowpack is a water equivalent of 11.7 cm (4.60 inches) [2-8]. This translates into a horizontal ground snow load of 1.15 kpa (24 pounds per square foot).

The greatest wind speed ever recorded in Boston was a one-minute sustained speed of 86 km per hour (54 miles per hour) with a peak gust of 130 km per hour (81 miles per hour) in March of 1993. There was no damage to the reactor building or nearby structures, including the cooling towers, from this storm.

Floods in the Boston/Cambridge area do not occur because of the drainage afforded by the Charles River. The most severe rains occurred in 1954 and 1955 during hurricanes. Roads that run parallel to the river, and whose under-passes are very close to the water table, did experience flooding. However, other areas, including the site of the reactor, were unaffected.

2.3.2 Site Meteorology

Atmospheric inversion data are shown in Table 2-4. This information was gathered at Logan Airport in Boston, Massachusetts. Stable conditions exist 16% of the time.

Wind speed and direction are continuously indicated in the Reactor Operations Office and on a strip chart recorder in the MITR Control Room. This equipment is located on the roof of the building adjacent to the MITR containment building at an elevation of approximately 24 meters (80 feet). If needed, other weather information is obtained from the Boston National Weather Service.

Table 2-4

Pasquill Stability Classes Observed at Logan Airport 1993-1995

Stability Class	Relative Frequency (%)
A - Extremely Unstable	0.08
B - Unstable	1.83
C - Slightly Unstable	8.30
D - Neutral	73.9
E - Slightly Stable	12.0
F - Extremely Stable	3.82

Source: U.S. Environmental Protection Agency [2-6].

2.4 Hydrology

Natural drainage from the reactor site is into the Charles River which flows into Boston Harbor and Massachusetts Bay. The mouth of the river is blocked by the Charles River Dam (Colonel Richard Gridley Locks). The primary purpose of this dam is to prevent salination of the river, which would in turn accelerate pollution. A consequence of this dam is that the Charles River Basin, which extends upstream from the dam, shows no tidal effects.

Flooding from rainfall or melting snow is not an issue in the Cambridge-Boston area because of the drainage provided by the river and also the size of the Charles River Basin. Minor flooding of two roads, both restricted to passenger vehicles, that parallel the river is a chronic problem but this does not affect the MITR. These are Memorial Drive and Storrow Drive, one on each bank. The underpasses for both roads are at or very nearly at the water table and minor flooding does occur during heavy downpours. This situation has no effect on the reactor site. Vehicles that would respond to an emergency at the MITR do not use either of these roads.

Seismically induced flooding is also unlikely. Any failure of man-made structures, such as Gridley Locks, would only serve to enhance the drainage away from the area. There are no flow restrictions on the Charles River upstream from the reactor site. Because of the absence of surface faulting in the area, there are no recorded instances of tsunami. Any instances of seismically induced "sea waves" would be too small to have any effect other than directly on the coastline. Diversion of the Charles River toward the MITR site is also unlikely because the site is slightly elevated in relation to the river.

The ground water level beneath the reactor site is at a depth of about eight feet below the surface. The land in the area, which was originally marsh, has been filled to a depth of ten to twelve feet. Thus, the ground water is in the fill.

2.5 Geology, Seismology, and Geotechnical Engineering

2.5.1 Regional Geology

The Cambridge-Boston metropolitan area is located within the Appalachian Mountain System which extends north to south from Newfoundland to Alabama and west to east from eastern New York State to the continental shelf. The Appalachian formation is quite old in geological time and has been subject to much erosion. The geologic history of the New England area can be summarized as follows:

a) Precambrian: The Precambrian era is defined as occurring more than 600 million years ago. The Canadian Shield area arose during this time during one of several episodes of mountain building. Not much else is known with any certainty from this era.

b) Paleozoic: The Paleozoic era is defined as occurring more than 230 million years ago but less than 600 million years ago. Tectonic activity including folding, faulting, and igneous intrusion occurred on at least two occasions (425 million or more years ago and 350 million or more years ago) to create the Appalachian Mountains.

c) Late Paleozoic and Mesozoic: The Mesozoic era is defined as occurring more than 63 million years ago but less than 230 million years ago. Additional tectonic activity occurred both 230 million or more years ago and 135 million or more years ago. These contributed to the creation of the Appalachian Mountains. However, they were less severe than the activities that occurred during the Paleozoic era.

d) Cenozoic: The Cenozoic era is defined as occurring less than 63 million years ago. This era was characterized by an absence of strong tectonic activity.

In more recent geologic time, glaciers covered the region on several occasions and then retreated. The last such occurrence was 10 to 15 thousand years ago. One result of this glacial activity was the basin in which Cambridge and Boston are located. The glaciers scraped the bedrock that now lies beneath Cambridge and Boston with the result that the depth of this bedrock is uneven.

There are several zones where seismic activity has been historically observed in the New England area even though the region as a whole is not considered a seismically-active one [2-9]. These are:

a) Near New York City.

b) Near the Mouth of the Connecticut River.

c) Near Narragansett Bay (Rhode Island).

d) An area comprising the middle of New Hampshire, Upstate New York, and Vermont.

Earthquakes have also occurred in historic times in the vicinity of Boston and Cambridge. These have tended to be either north or south of the two cities but not in the cities themselves. Figure 2-7, which is from a study by Cornell, Asce, and Merz who were funded by the National Science Foundation, shows these areas [2-9]. The seismic events shown all had a Modified Mercalli Intensity (MMI) of V or more.

2.5.2 Site Geology

MIT recently had studies conducted of the geology of the University Park site which is in very close proximity to the reactor site [2-10]. The principal findings were:

a) The site is underlain by a shale-type rock known as the "Cambridge Argillite Member." Granite, which is harder, is found to the north and south.

b) The starting depth of the bedrock below the surface is variable because of the action of the glaciers.

c) A layer of clay ("Boston Blue Clay") overlays the bedrock. This clay was the result of sedimentation when, after the glacial retreat, the area was submerged. The clay layer is 75 to 100 feet thick.

d) Sand and gravel overlie the clay. This is in turn covered by organic material and fill. Table 2-5 summarizes the soil stratification at University Park.

A unique feature of the Boston Blue Clay is its rather high ion exchange capacity which varies from 5 to 15 milli-equivalents per 100 grams. This suggests that if contaminants were washed into the soil, they would become fixed in the clay. It is also worth noting that soil samples taken at the MITR site in 1999 showed no radioactivity from operation of the reactor facility. These tests included a check for tritium.

Table 2-5

Soil Stratification

Type	Thickness
Fill	8.5' - 10'
Organic	2.5' - 7.75'
Sand	~10'
Silt	3' to 5'
Clay	75' - 100'
Bedrock	N/A

Source: David V. Lewin Corp./Geotechnical Engineering [2-10].

2.5.3 Seismicity

Appendix A to this chapter of this report is a listing of all historically reported earthquakes within a 200 km radius of the reactor site with an MMI of greater than IV or a Richter magnitude greater than 3.0. It should be noted that the New England area has one of the longest histories of reporting of earthquake activities in the United States dating back to the 16th century. Hence, the length of this listing appears large, despite the relative infrequency of earthquakes in the area.

2.5.4 Maximum Earthquake Potential

The largest earthquake every recorded in New England was the Cape Ann earthquake on November 18, 1755. It is estimated to have had an epicentral intensity of VIII on the MMI scale. Seismologists at the Weston Observatory put the intensity in Boston from the Cape Ann event at a maximum (poor soil) value of VII on the MMI scale. See Figure 2-8.

The risk of a major seismic event in the Boston area has been studied by Cornell, Asce, and Merz [2-9] and by Borosh et al. [2-11]. Among the findings of these studies:

a) The correlation between post-seismic events and known geotectonic structure is weak for the Boston area [2-9].

b) Surface faulting has never been observed for an earthquake in the region [2-11].

c) It can be inferred that for an earthquake in the Boston area, the "strong motion duration will be relatively short, about 5 seconds and the relative frequency content will emphasize higher frequencies. This latter condition is interpreted to mean that one might expect a peak ground velocity of about 3.6 inches per second with any motion whose peak ground acceleration was 0.1 g" [2-9].

The above findings support the choice of a maximum safe shutdown earthquake acceleration of 0.15 g which is used in Chapter 13 of this report in the analysis of the building's response to seismic events. This figure is also the one selected by the Pilgrim Nuclear Power Station that is located about 60 miles to the southeast of the MITR site.

Cornell, Asce, and Merz have made an estimate of the annual risk of an event that equals or exceeds a given intensity versus intensity for Boston. This is shown in Figure 2-9. It gives a most likely mean return period of 10,000 years for an earthquake of MMI-VII and about 700 years for one of MMI-VI.

2.5.5 Surface Faulting

Surface faulting has not been observed for an earthquake in the region [2-11].

2.5.6 Liquefaction Potential

The study performed of the University Park area for MIT addressed this issue [2-10]. It recommended removal of the fill and organic material. (Note: This type of material was also removed at the reactor site prior to construction in 1957-1958.) The study thus concluded, "Based on Massachusetts State Building Code considerations for earthquake and lateral load, this is a Class B soil site and a factor of S=1.5 is indicated for use in determining lateral forces. The sand stratum is not considered to be susceptible to liquefaction." Upon application of the factor of 1.5, the maximum safe shutdown earthquake acceleration becomes 0.225 g.

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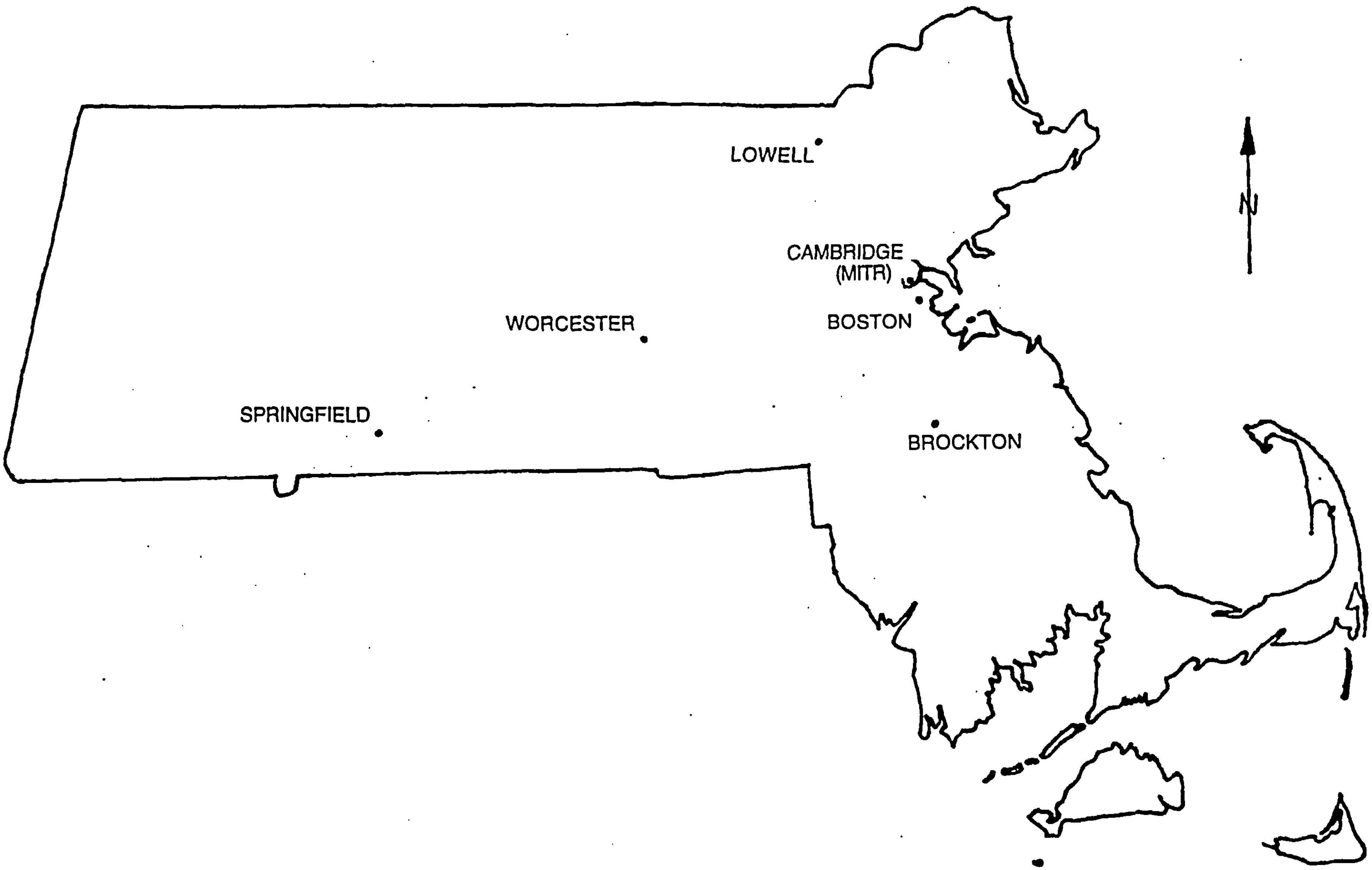


FIGURE 2-1 MASSACHUSETTS

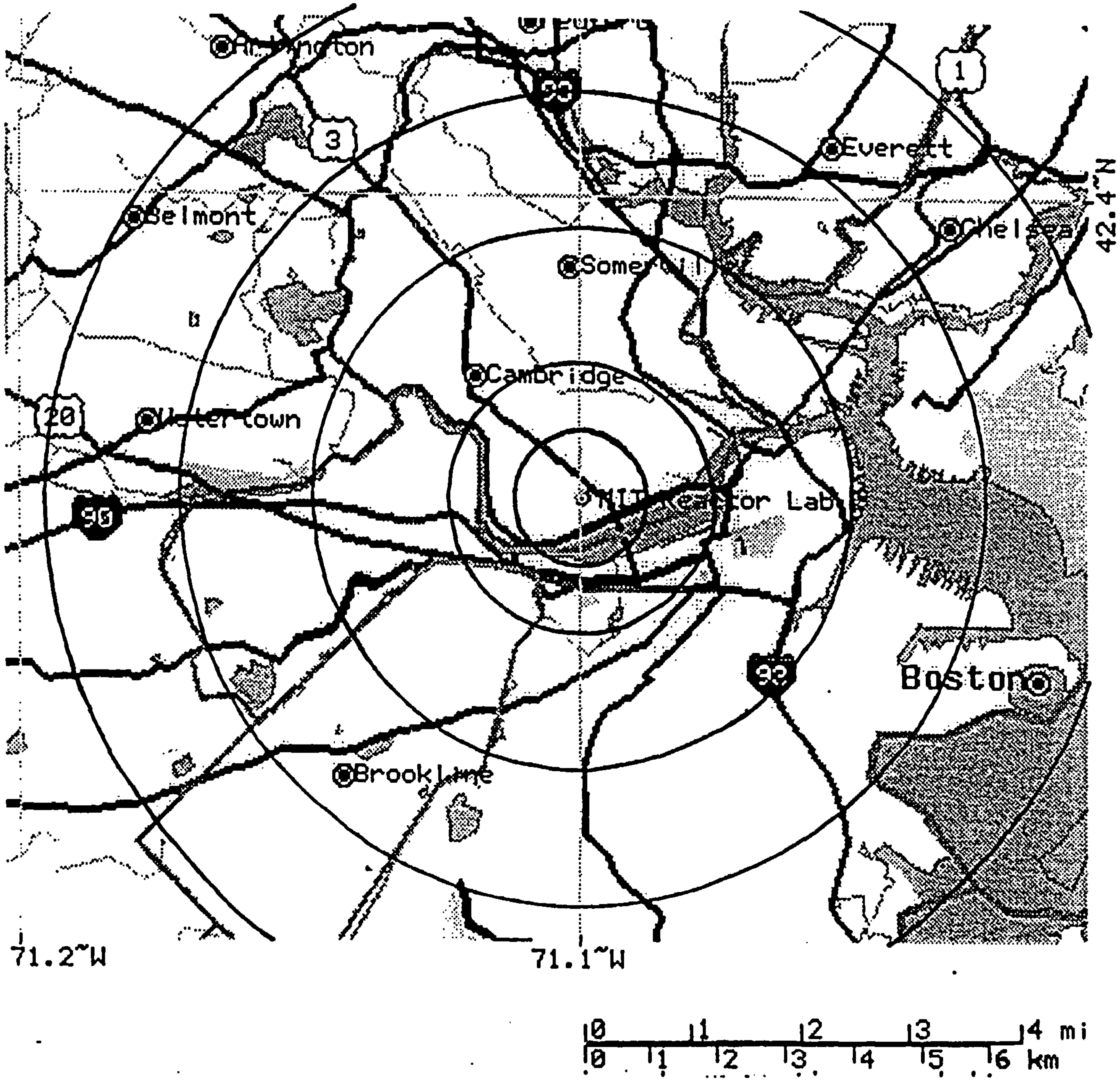


FIGURE 2-2 GREATER BOSTON AREA

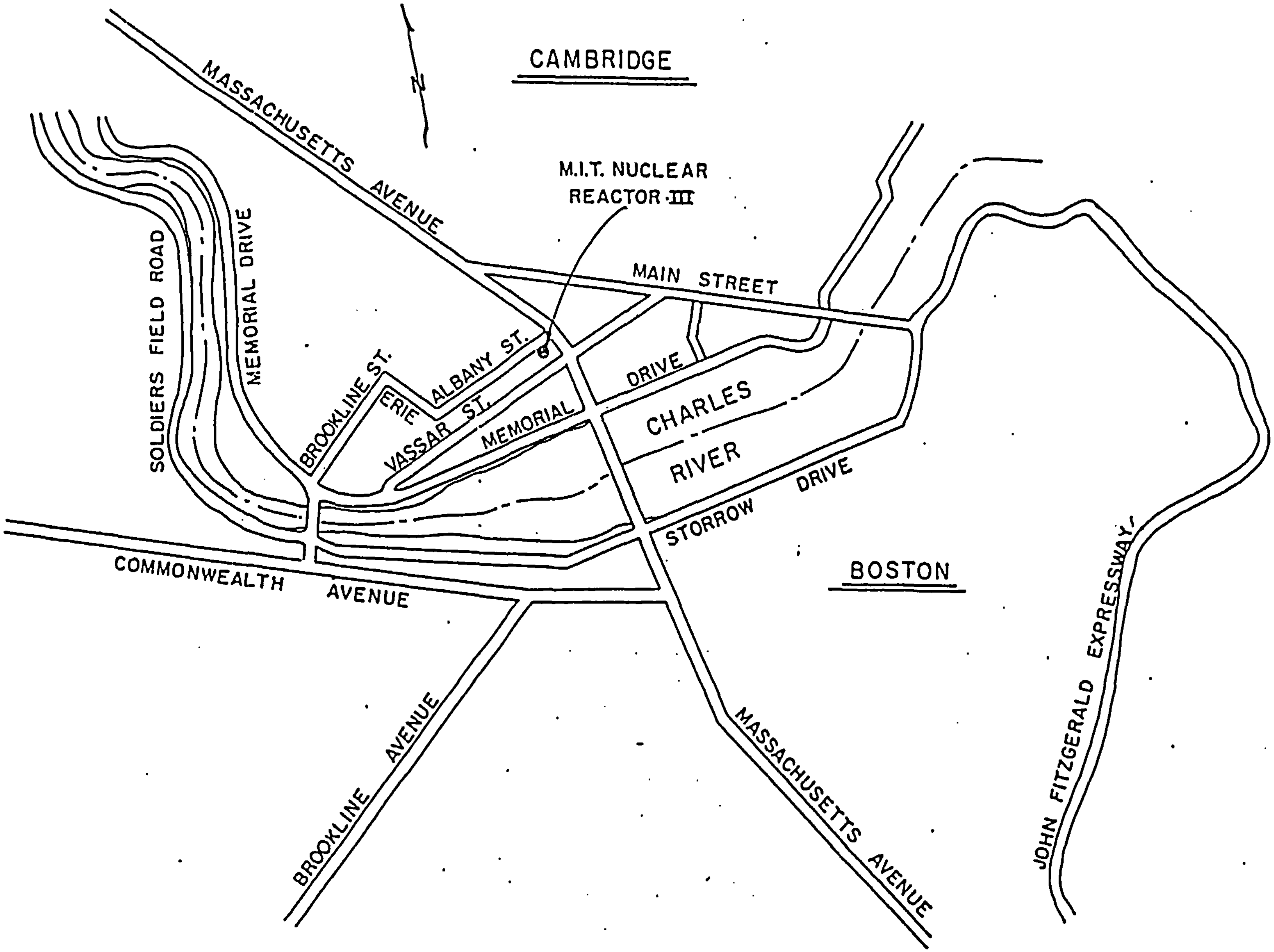


FIGURE 2-3 CHARLES RIVER BASIN

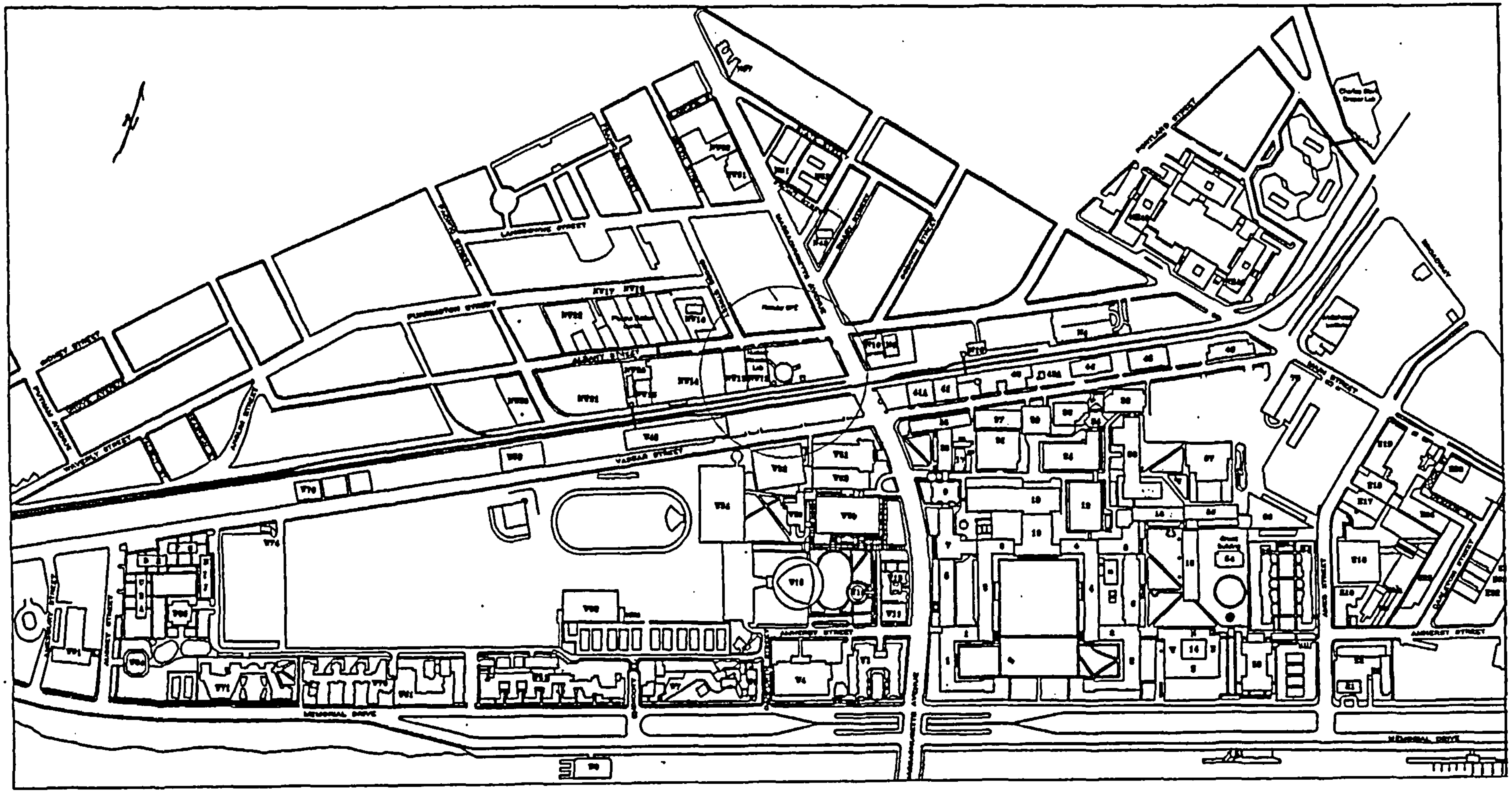


FIGURE 2-4 MIT CAMPUS

FIGURE 2-5 REACTOR OPERATIONS, RESTRICTED AREA, & SITE BOUNDARIES

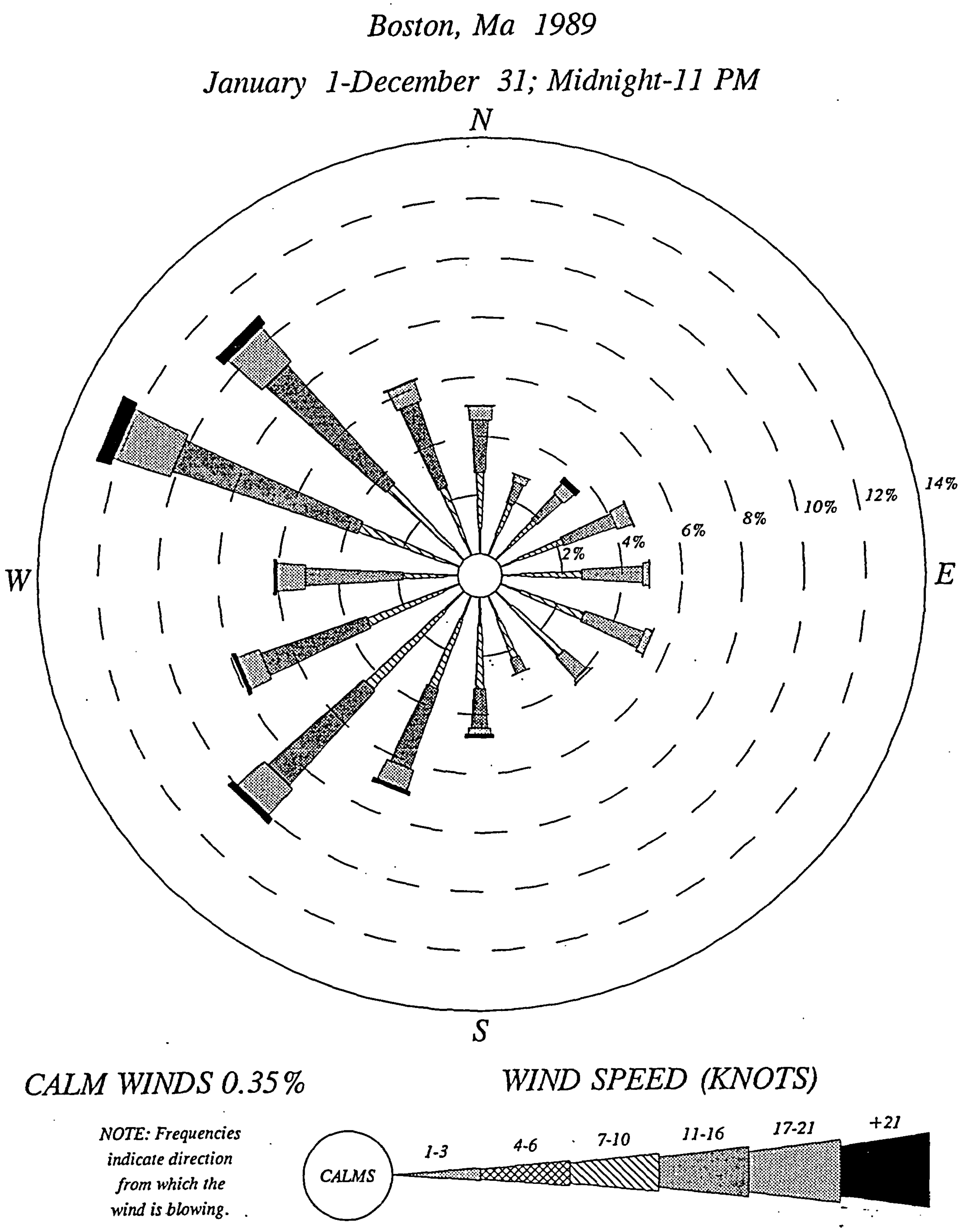
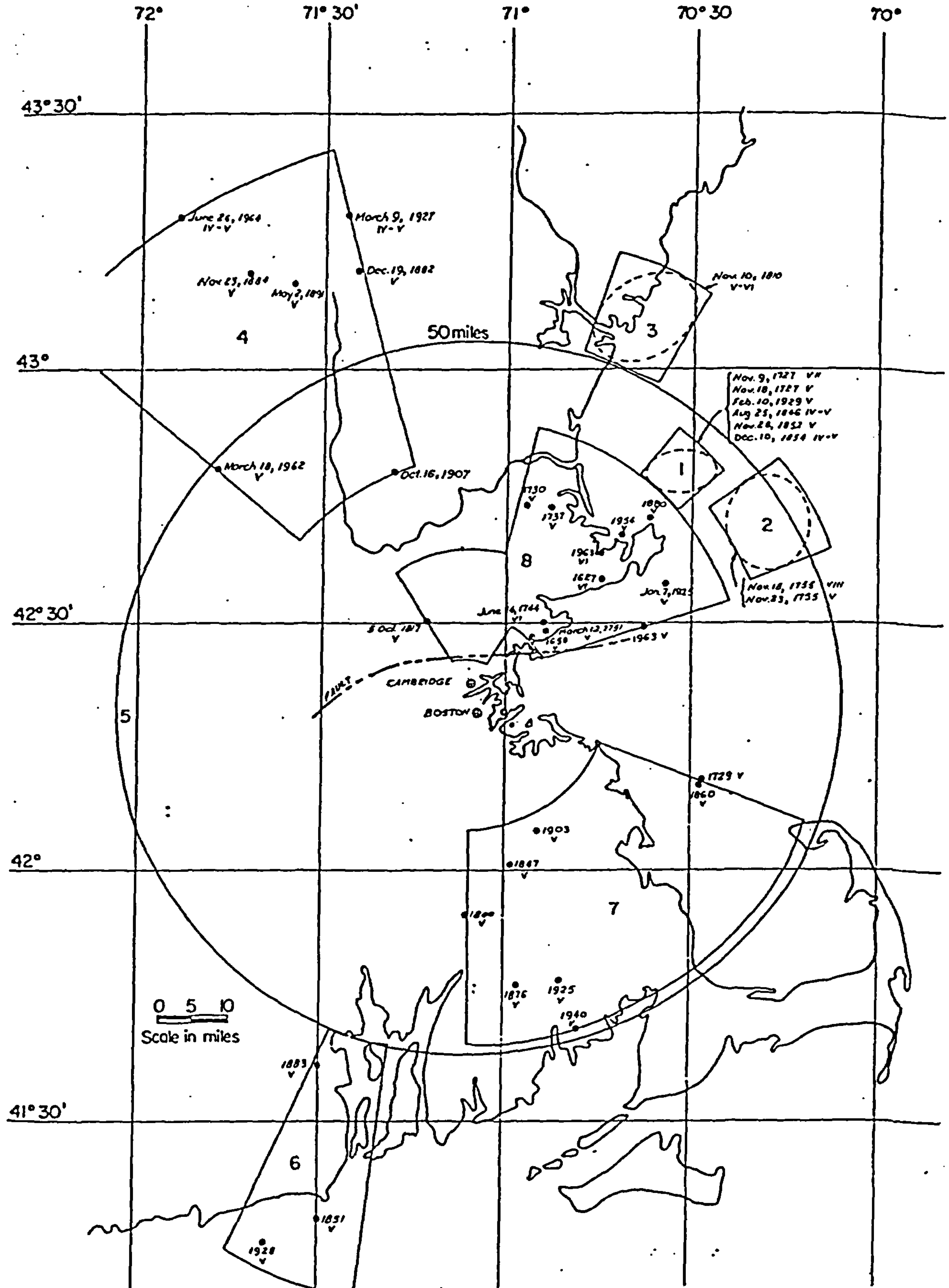


FIGURE 2-6 BOSTON LOGAN AIRPORT WIND ROSE



Source: Cornell, Asce, and Merz [2-9]

FIGURE 2-7 HISTORICAL EVENTS, MMI > V, SHOWING GROSS SOURCES

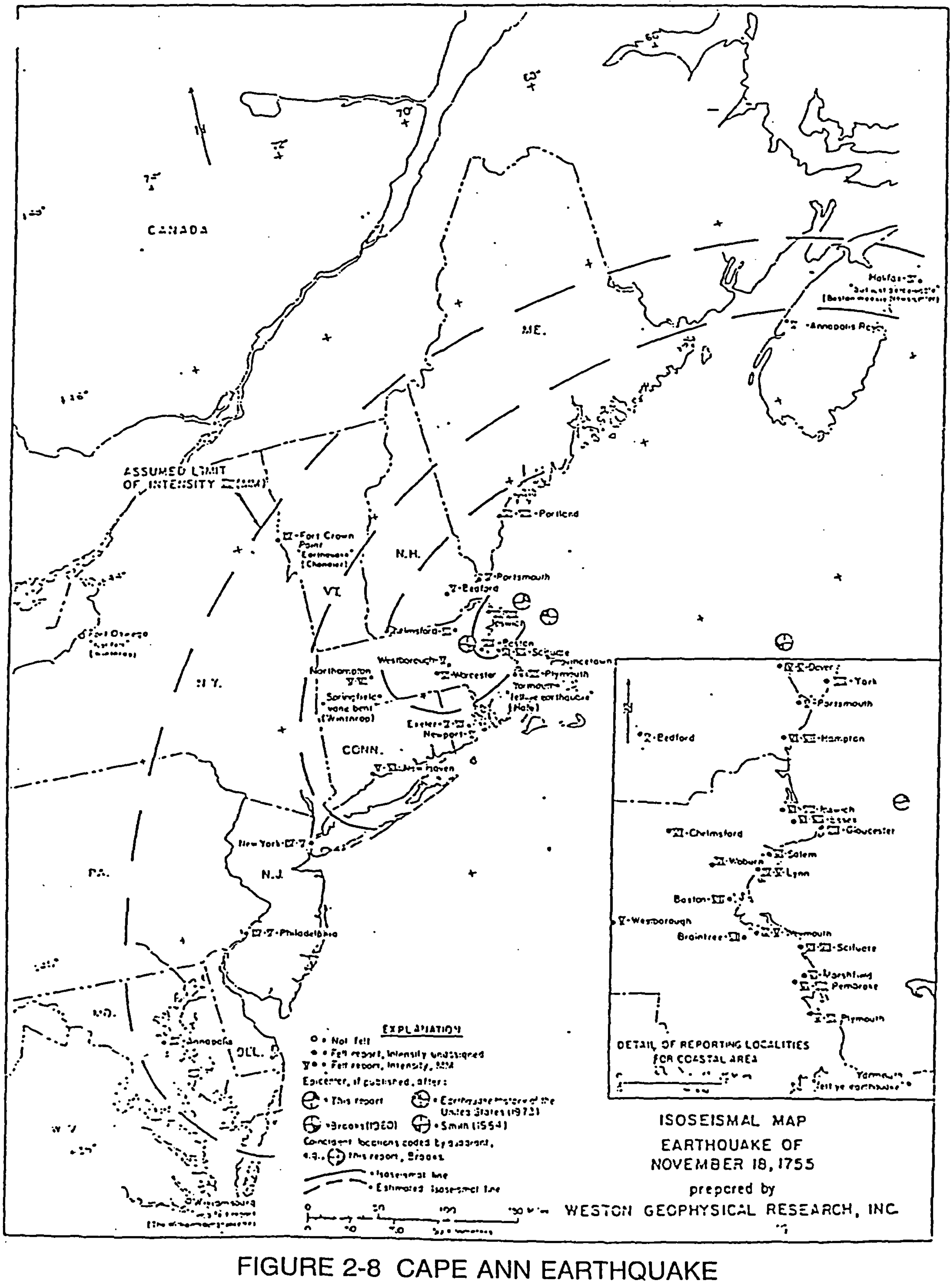
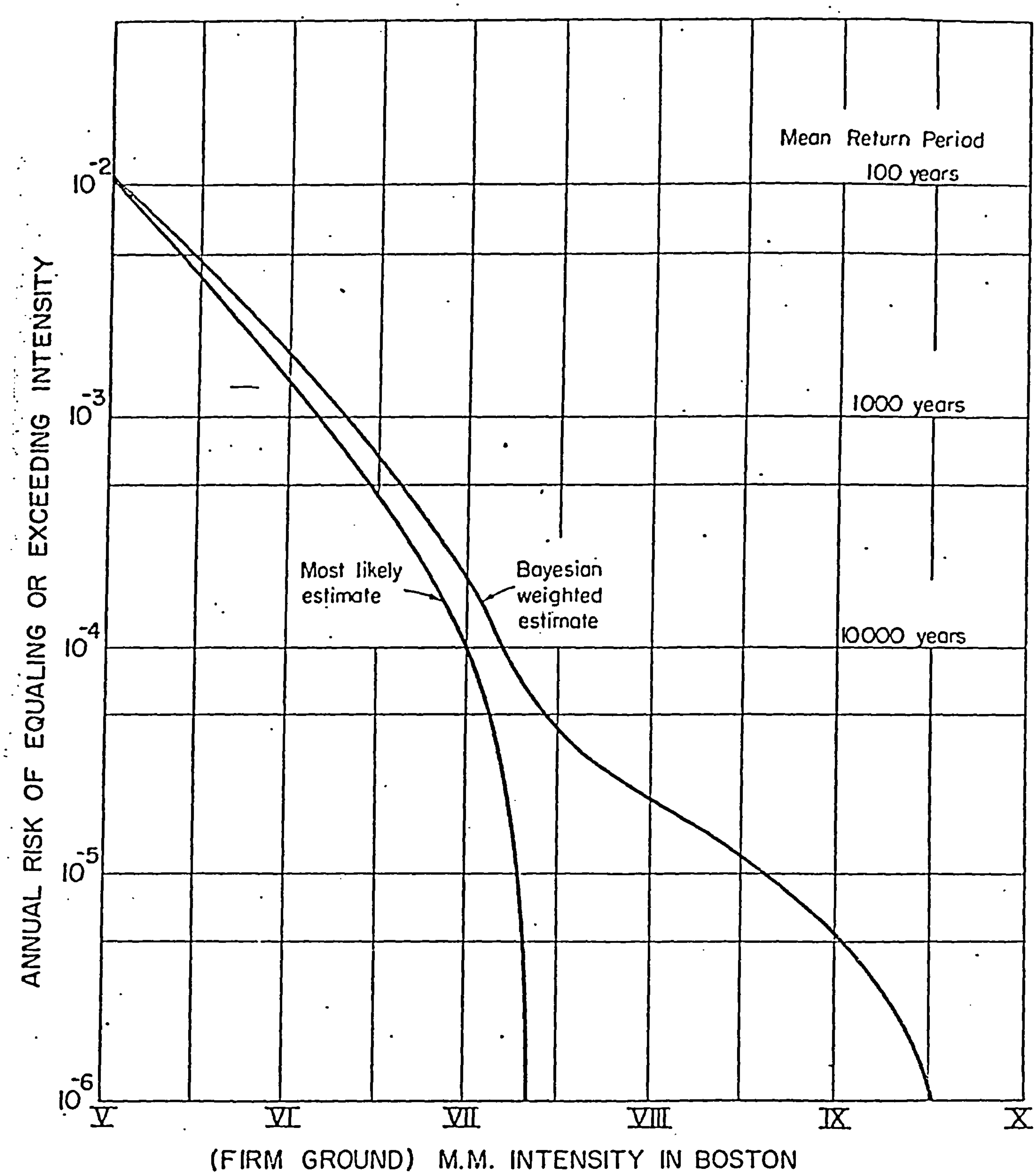


FIGURE 2-8 CAPE ANN EARTHQUAKE



Source: Cornell, Asce, and Merz [2-9]

FIGURE 2-9 RISK ESTIMATE BY INTENSITY

Appendix A

Summary of Seismic History for New England

Regional Earthquake History (Refs. 2-12, 2-13)

Date	Time	North Latitude	West Longitude	MM Intensity	Richter Magnitude	Location
1568		41.5	72.5	VI		Moodus - E. Haddam, CT
1574		41.5	72.5	V		Moodus - E. Haddam, CT
1584		41.5	72.5	V		Moodus - E. Haddam, CT
1592		41.5	72.5	V		Moodus - E. Haddam, CT
1627		42.6	70.8	VI		Essex, MA
03/15/1643	1200	42.8	70.8	IV		Newbury, MA
06/11/1643	1800	42.8	70.8	IV		Newbury, MA
11/08/1653		42.6	70.9	IV		Danvers, MA
04/14/1658		42.5	70.9	V		Lynn, MA
04/03/1668	1400	42.3	71.1	IV		Boston, MA
12/13/1677		41.1	73.5	IV		Stamford, CT
02/18/1685		42.7	70.8	IV		Danvers, MA
1698		41.4	73.5	IV		Danbury, CT
1702		42.4	71.1	IV		Danbury, CT
06/27/1705		42.4	71.1	IV		Boston, MA
1711		41.4	73.5	IV		Danbury, CT
11/10/1727	0340	42.8	70.6	VII		Cape Ann, MA
11/10/1727	0435	42.8	70.6	IV		Cape Ann, MA
11/10/1727	0715	42.8	70.6	IV		Cape Ann, MA
11/14/1717	2200	42.8	70.6	V		Cape Ann, MA
11/18/1727	1620	42.8	70.6	IV		Cape Ann, MA
11/24/1727	1000	42.8	70.6	IV		Cape Ann, MA
12/01/1727		42.8	70.6	IV		Cape Ann, MA
12/16/1727		42.8	70.6	IV		Cape Ann, MA

Date	Time	North Latitude	West Longitude	MM Intensity	Richter Magnitude	Location
12/19/1727	1500	42.8	70.6	IV		Cape Ann, MA
12/29/1727	0330	42.8	70.6	IV		Cape Ann, MA
01/05/1728	0300	42.8	70.6	IV		Cape Ann, MA
01/18/1728	0200	42.8	70.6	IV		Cape Ann, MA
02/05/1728	0230	42.8	70.6	IV		Cape Ann, MA
02/08/1728	1130	42.8	70.6	IV		Cape Ann, MA
02/10/1728	2030	42.8	70.6	V		Cape Ann, MA
05/16/1728		42.8	70.6	IV		Cape Ann, MA
05/24/1728	0240	42.8	70.6	IV		Cape Ann, MA
05/29/1728	0100	42.8	70.6	IV		Cape Ann, MA
06/02/1728	1500	42.8	70.6	IV		Cape Ann, MA
07/30/1728	1500	42.8	70.6	IV		Cape Ann, MA
08/02/1728	0315	42.8	70.6	IV		Cape Ann, MA
02/10/1729	1400	42.8	70.6	V		Cape Ann, MA
03/30/1729	1900	42.8	70.6	IV		Cape Ann, MA
09/19/1729	2030	42.8	70.6	IV		Cape Ann, MA
10/10/1729	2130	42.8	70.6	IV		Cape Ann, MA
11/25/1729	1300	42.8	70.6	IV		Cape Ann, MA
12/09/1729	0100	42.8	70.6	IV		Cape Ann, MA
02/20/1730	0100	42.8	70.6	IV		Cape Ann, MA
02/20/1730	0500	42.8	70.6	IV		Cape Ann, MA
03/09/1730	1845	42.8	70.6	IV		Cape Ann, MA
04/24/1730	0100	42.8	70.6	IV		Cape Ann, MA
12/07/1730	0120	42.8	70.6	IV		Cape Ann, MA
12/24/1730	0330	42.8	70.6	IV		Cape Ann, MA
01/13/1731	0000	42.8	70.6	IV		Cape Ann, MA

Date	Time	North Latitude	West Longitude	MM Intensity	Richter Magnitude	Location
01/19/1731	0000	42.8	70.6	IV		Cape Ann, MA
01/23/1731	0500	42.8	70.6	IV		Cape Ann, MA
07/16/1731	1000	42.8	70.6	IV		Cape Ann, MA
10/13/1731	0400	42.8	70.6	IV		Cape Ann, MA
02/19/1732	0000	42.8	70.6	IV		Cape Ann, MA
11/23/1734	0500	42.8	70.6	IV		Cape Ann, MA
02/13/1736	2245	42.8	70.6	IV		Cape Ann, MA
10/12/1736	0630	42.8	70.6	IV		Cape Ann, MA
11/23/1736	0700	42.8	70.6	IV		Cape Ann, MA
02/17/1737	2115	42.8	70.6	IV		Cape Ann, MA
09/20/1737	1520	42.8	70.6	IV		Cape Ann, MA
08/13/1739	0730	42.8	70.6	IV		Cape Ann, MA
02/05/1741	2050	42.8	70.6	IV		Cape Ann, MA
12/17/1741	1300	42.3	71.2	IV		Boston, MA
06/14/1744	1515	42.5	70.9	VI		Cape Ann, MA
06/14/1744	2200	42.5	70.9	IV		Salem, MA
07/01/1744		42.5	70.9	IV		Salem, MA
11/18/1755	0912	42.7	70.3	VIII		Off Cape Ann, MA
11/18/1755	1029	42.7	70.3	IV		Off Cape Ann, MA
11/23/1755	0127	42.7	70.3	V		Off Cape Ann, MA
02/20/1755	0115	42.7	70.3	IV		Off Cape Ann, MA
08/08/1757	1915	42.3	71.1	IV		Boston, MA
02/02/1759	0700	42.3	71.0	IV		Boston, MA
03/12/1761	0715	42.5	70.9	V		Boston, MA
03/16/1761		42.3	71.1	IV		Boston, MA
11/02/1761	0100	43.1	71.5	IV		S. of Concord, NH

Date	Time	North Latitude	West Longitude	MM Intensity	Richter Magnitude	Location
01/23/1766	1000	43.7	70.3	IV		Portland, ME
02/02/1766		42.0	68.0	VI		Off Cape Cod, MA
08/25/1766		41.5	71.3	IV		Newport, RI
12/17/1766	1148	43.1	70.8	IV		Portsmouth, NH
10/19/1769	AM	43.7	70.3	IV		Portland, ME
10/19/1769	1700	43.7	70.3	IV		Portland, ME
11/29/1780		42.5	70.9	IV		Lynn, MA
05/16/1791	1300	41.5	72.5	VI		Moodus - E. Haddam, CT
08/29/1792	0300	41.5	72.5	IV		Moodus - E. Haddam, CT
10/24/1792	0600	41.5	72.5	IV		Moodus - E. Haddam, CT
01/11/1793	1300	41.5	72.5	IV		Moodus - E. Haddam, CT
07/06/1793	1100	41.5	72.5	IV		Moodus - E. Haddam, CT
03/06/1794	1900	41.5	72.5	IV		Moodus - E. Haddam, CT
03/07/1794	0400	41.5	72.5	IV		Moodus - E. Haddam, CT
03/09/1794	1900	41.5	72.5	IV		Moodus - E. Haddam, CT
03/10/1794	0400	41.5	72.5	IV		Moodus - E. Haddam, CT
12/20/1800		43.7	72.3	IV		NW of Newport, NH
12/25/1800		41.9	71.1	IV		Wareham-Taunton, MA
03/01/1801	2030	43.1	70.8	IV		Portsmouth, NH
01/18/1803	1450	42.5	70.9	IV		Salem, MA
04/06/1805	1915	42.5	70.9	IV		Lynn, MA
04/25/1805		42.5	70.9	IV		Salem, MA
01/14/1807	0400	43.0	71.1	IV		Near Exeter, NH
05/06/1807	1800	43.5	70.5	IV		Saco River, ME
11/10/1810	0215	43.0	70.8	V		Portsmouth, NH
11/09/1814	0014	43.7	70.3	V		Windham, ME

Date	Time	North Latitude	West Longitude	MM Intensity	Richter Magnitude	Location
10/05/1817	1645	42.5	71.2	VI		Woburn, MA
03/07/1823	1500	43.9	70.0	IV		Brunswick, ME
08/23/1827		41.4	72.7	IV		NW of New London, CT
07/25/1828	1100	43.9	70.0	IV		Brunswick, ME
01/01/1829		43.1	70.8	IV		Portsmouth, NH
08/27/1829	2145	44.2	69.8	IV		Gardiner, ME
01/15/1837	0700	42.5	70.9	IV		Lynn, MA
04/12/1837		41.7	72.7	V		Hartford, CT
08/09/1840	2030	41.5	72.9	V		Hartford, CT
10/24/1843		41.1	71.2	IV		Canton, MA
10/26/1845	2315	41.2	73.3	V		Bridgeport, CT
05/30/1846	1830	42.7	70.3	IV		Cape Ann, MA
08/25/1846	0945	42.5	70.8	V		Marblehead, MA
08/08/1847	1500	41.7	70.1	VI		Brewster, MA
10/08/1849	PM	42.5	71.4	IV		Middlesex Co., MA
01/10/1852	1140	41.2	71.4	IV		Off Coast, RI
11/28/1852	0445	43.0	70.9	V		Exeter, NH
07/17/1853	1030	43.5	70.2	IV		Portland, ME
11/28/1853		43.0	71.9	IV		Antrim, NH
11/28/1853	0445	43.0	69.0	IV		Cape Ann, MA
10/25/1854	0300	42.9	72.3	IV		Keene, NH
12/11/1854	1530	43.0	70.8	V		North Hampton, NH
01/16/1855	2300	44.0	71.0	V		Otisfield, ME
01/17/1855	0020	44.0	71.0	IV		Otisfield, ME
02/19/1855		44.6	69.6	IV		Ellsworth, ME
05/29/1855	1000	44.7	71.6	IV		Coos Co., NH

Date	Time	North Latitude	West Longitude	MM Intensity	Richter Magnitude	Location
03/13/1856	0300	41.4	72.6	IV		Haddam, CT
07/01/1857	0345	41.5	72.5	IV		Moodus - E. Haddam, CT
12/23/1857	1830	44.1	70.2	VI		Lewiston, ME
12/28/1857		44.1	70.2	IV		Lewiston, ME
06/27/1858		41.4	72.8	IV		North Haven, CT
07/01/1858	0345	41.3	73.0	IV		New Haven, CT
03/17/1860	0230	42.2	70.5	IV		Off Provincetown, MA
03/17/1860	0315	42.2	70.5	IV		Off Provincetown, MA
02/03/1862	0100	41.5	72.5	IV		Moodus - E. Haddam, CT
02/08/1870		44.1	69.8	IV		Gardiner, ME
07/20/1871		43.2	71.5	IV		Concord, NH
11/18/1872	1900	43.2	71.6	V		Concord, NH
01/06/1874		43.6	71.2	IV		Wolfeboro, NH
01/25/1874	1700	42.6	71.4	IV		Lowell, MA
01/26/1874	0700	43.0	71.5	IV		Manchester, NH
11/24/1874		42.7	70.9	IV		Salem-Newbury, MA
07/28/1875	0910	41.9	73.0	V		NW of Torrington, CT
12/01/1875	0900	42.9	72.3	IV		Keene, NH
08/22/1876	0430	41.5	71.3	V		Newport, RI
10/26/1879	0330	43.0	71.5	IV		Manchester, NH
05/12/1880	1245	42.7	71.0	V		Boxford, MA
01/21/1881	0240	44.0	70.0	IV		Bath, ME
06/19/1881	0825	42.8	70.9	IV		Newbury, MA
04/17/1882	1900	43.2	71.7	IV		Hopkinton, NH
02/04/1883	2005	43.6	71.2	IV		Wolfeboro, NH
02/28/1883	0330	41.5	71.3	V		Newport, RI

Date	Time	North Latitude	West Longitude	MM Intensity	Richter Magnitude	Location
01/06/1886	0010	42.9	71.5	IV		Merrimack, NH
01/17/1886	2214	42.8	71.4	IV		Nashua, NH
08/05/1886		41.5	72.5	IV		Moodus - E. Haddam, CT
07/01/1887	0200	43.2	71.5	IV		Concord, NH
02/01/1888	1620	44.7	70.1	IV		Industry, ME
03/08/1889		43.5	71.6	IV		Franklin, NH
05/02/1891	0010	43.2	71.6	V		Near Concord, NH
05/30/1891	0000	43.1	71.5	IV		Near Concord, NH
12/11/1992	1630	44.3	71.7	IV		Bethlehem, NH
04/10/1894	AM	41.6	72.5	IV		Moodus - E. Haddam, CT
10/22/1896	1030	44.3	71.8	IV		Bethlehem, NH
07/01/1897	0920	43.7	71.6	IV		Meredith, NH
08/05/1897		41.5	72.5	IV		Moodus - E. Haddam
06/11/1898	0645	42.8	72.6	IV		Brattleboro-Vernon, VT
05/17/1899	0115	41.6	72.6	V		Moodus - E. Haddam, CT
01/21/1903	AM	42.1	70.9	V		Whitman, MA
01/22/1903		42.0	71.3	IV		Attleboro, MA
04/24/1903	1230	42.7	71.0	IV		Merrimac Valley, MA
05/27/1905		44.3	72.6	II		Montpelier, VT
08/30/1905	1040	43.1	70.7	V		Rockingham Co., NH
11/26/1905	0030	41.5	71.3	IV		Newport, RI
05/08/1906	1330	41.5	72.5	IV		Moodus - E. Haddam, CT
10/20/1906	1415	43.5	70.5	IV		Saco, ME
06/29/1907		43.5	70.5	IV		Biddeford, ME
10/16/1907	0010	42.8	71.0	V		Newbury, MA
11/23/1908	1300	43.5	717	IV		Franklin, NH

Date	Time	North Latitude	West Longitude	MM Intensity	Richter Magnitude	Location
01/23/1910	0130	43.8	70.4	IV		Windham, ME
08/21/1910	1845	42.7	71.1	IV		Merrimac Valley, MA
08/30/1910	1430	43.4	72.1	IV		Lake Sunapeo, NH
03/02/1911	2130	43.2	71.5	IV		Concord, NH
11/03/1913	1430	41.5	71.5	IV		Kingstown, RI
01/13/1914	0800	45.2	67.3	IV		Calais, ME
02/21/1915	0203	42.8	71.1	IV		Merrimac Valley, MA
02/16/1917	0900	41.5	72.5	IV		Moodus - E. Haddam, CT
07/11/1919	0140	43.9	70.0	IV		Brunswick, ME
07/23/1919	1150	43.7	70.3	IV		Portland, ME
05/23/1920	0800	43.1	71.5	IV		Concord, NH
07/07/1920	0800	43.5	70.5	IV		Saco, ME
07/29/1921	2114	42.5	70.4	IV		Cambridge, MA
05/07/1922	2240	43.4	71.4	IV		Pittsfield, NH
03/09/1925		42.9	71.5	IV		Goff's Falls, NH
04/24/1925	0756	41.7	70.8	V		Wareham, MA
05/04/1925	1751	42.5	70.9	IV		Lynn, MA
10/09/1925	1355	43.7	71.1	VI		Ossipee, NH
10/30/1925	AM	41.5	72.5	IV		Moodus - E. Haddam, CT
11/14/1925	1304	41.7	72.4	V		N. of Hebron, CT
11/16/1925	0620	41.8	72.7	IV		Hartford, CT
01/04/1926		41.6	71.8	IV		Voluntown, CT
03/18/1926	2109	42.8	71.8	V		New Ipswich, NH
08/28/1926	2130	44.8	70.4	IV		Farmington, ME
03/09/1927	0408	43.3	71.4	IV		Concord, NH
03/30/1927	PM	41.7	72.8	IV		New Britain, CT

Date	Time	North Latitude	West Longitude	MM Intensity	Richter Magnitude	Location
08/20/1927		42.3	71.0	IV		Quincy, MA
04/25/1928	2338	44.5	71.2	V		Berlin, NH
04/28/1928	2207	43.2	71.5	IV		Concord, NH
02/05/1929	1909	44.0	70.3	IV		Auburn, ME
02/14/1930	0615	43.4	71.7	IV		Franklin, NH
03/19/1930	0015	43.3	71.6	IV		Concord, NH
01/17/1933	0530	41.6	70.9	IV		New Bedford, MA
01/30/1934	1030	41.8	72.6	IV		S. Windsor, CT
08/02/1934	1458	42.6	70.7	IV		Cape Ann, MA
08/02/1934	1459	43.7	70.3	IV		Portland, ME
08/03/1934	0230	43.7	70.3	IV		Portland, ME
04/24/1935	0124	42.2	70.2	IV		Off Cape Cod, MA
11/10/1936	0246	43.6	71.4	V		Laconia, NH
07/27/1937	0910	41.8	72.4	IV		Manchester, CT
06/23/1938	0357	42.6	71.4	IV		Chelmsford, MA
01/28/1940	2311	41.6	70.8	V	2.6	Buzzards Bay, MA
12/20/1940	0727	43.8	71.3	VII	5.8	Ossipee, NH
12/24/1940	1343	43.8	71.3	VII	5.8	Ossipee, NH
12/25/1940	0503	43.8	71.3	IV	4.0	Ossipee, NH
12/27/1940	1956	43.8	71.3	IV	3.9	Ossipee, NH
01/21/1941	0227	43.8	71.3	IV	3.6	Ossipee, NH
10/11/1941	0815	42.3	72.3	IV	3.0	Sturbridge, MA
03/14/1943	1402	43.7	71.6		3.9	Meredith, NH
12/14/1944	0314	41.6	72.8	IV		Meriden, CT
04/17/1949	0015	41.6	71.5	IV		N. Kingston, RI
01/26/1951	0327	41.5	72.5	IV		Moodus - E. Haddam, CT

Date	Time	North Latitude	West Longitude	MM Intensity	Richter Magnitude	Location
03/31/1951	0350	42.2	72.2	IV		Palmer, MA
06/10/1951	1720	41.5	71.5	IV	4.6	Kingstown, RI
05/11/1953	0613	44.0	71.1	IV		Conway, NH
02/13/1954		42.2	72.6	IV		Springfield, MA
02/13/1954		42.2	72.6	IV		Springfield, MA
07/29/1954	1956	42.7	70.7	V	4.0	Cape Ann, MA
04/26/1957	1140	43.6	69.8	VI	4.7	Portland, ME
09/19/1958	1745	43.6	70.2	V		Cape Elizabeth, ME
11/21/1958	2330	44.0	71.7	IV		Woodstock, NH
12/29/1962	0619	42.8	71.7	V		Nashua, NH
10/16/1963	1531	42.5	70.4	V	4.3	Marblehead, MA
10/30/1963	2236	42.7	70.8	IV	3.2	Off Cape Ann, MA
12/04/1963	2132	43.7	71.4	IV	3.6	Laconia, NH
04/01/1964	1121	43.4	71.5	IV	2.4	Laconia, NH
06/26/1964	1104	43.3	71.5	V	3.5	Concord, NH
01/03/1965	1705	43.5	71.5	III	3.4	Laconia, NH
10/24/1965	1745	41.3	70.1	V		Nantucket, MA
12/08/1965	0302	41.7	71.4	IV		Warwick, RI
04/28/1966	1202	44.1	71.9	IV		Benton, NH
10/23/1966	2305	43.0	71.4	IV	2.7	Manchester, NH
02/02/1967	1340	41.6	71.2	V	3.1	Narragansett Bay, RI
11/03/1968	0833	41.4	72.5	V	3.3	Moodus - E. Haddam, CT
08/06/1969	1602	43.8	71.4	V	2.6	Ossipee, NH
09/19/1970	1335	42.9	71.9	IV	2.6	Greenfield, NH
10/21/1971	0054	42.7	71.2	V	2.3	Lawrence, MA
02/26/1973	1342	44.5	70.0		3.0	Belgrade Lakes, ME

Date	Time	North Latitude	West Longitude	MM Intensity	Richter Magnitude	Location
12/22/1974	2046	42.4	69.8		3.0	NE of Provincetown, MA
05/10/1976	0134	41.5	71.0	IV	2.7	New Bedford, MA
12/20/1977	1744	41.8	70.7	IV	3.1	Wareham, MA
12/25/1977	1535	43.2	71.7	IV	3.2	Hopkinton, NH
01/04/1978	1928	44.0	70.5		3.2	Lewiston, ME
07/28/1979	2329	43.3	70.3		3.5	Portland, ME
10/24/1980	1727	41.3	72.9		3.1	New Haven, CT
10/25/1980	0412	41.3	72.8		3.0	New Haven, CT
06/28/1981	2242	43.5	71.6		3.0	Laconia, NH
10/21/1981	1649	41.2	72.5		3.8	Moodus - E. Haddam, CT
01/19/1982	0144	43.5	71.6		4.7	Laconia, NH
01/27/1982	1850	41.9	71.0		3.0	Middleboro, MA
06/17/1982	1414	41.5	73.0		3.0	Windsor, CT
10/15/1985	2000	42.5	71.5		3.0	Littleton, MA
10/25/1986	1716	43.4	71.6		3.9	Laconia, NH
08/29/1989	1556	41.6	70.9		3.0	New Bedford, MA
10/06/1992	1536	43.3	71.6		3.4	Laconia, NH
10/02/1994	1127	42.3	72.3		3.5	Palmer, MA
10/02/1994	1436	42.3	72.2		3.1	Palmer, MA
03/20/1996	2022	41.8	71.2		3.5	Middleboro, MA

Chapter 3

Design of Structures, Systems, and Components

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Chapter 3

Design of Structures, Systems, and Components

3.1 Design Criteria

Design criteria for various components and systems of the MITR are summarized here. Descriptions of these components and systems are deferred to the cognizant chapter of this report.

3.1.1 Prevention of Release of Radioactive Material

Operation of the MITR results in the production of fission products in the fuel itself, tritium in the heavy-water reflector, and byproduct material. The latter is produced by both core component activation and sample irradiation. There are multiple barriers to the release of these radioactive materials. These are:

a) Fission Products from the Core: The fuel matrix itself, the fuel clad, the light-water core tank and primary coolant system, and the containment building.

b) Fission Products from the Fission Converter: The fuel matrix itself, the fuel clad, the fission converter tank and associated coolant system, and the containment building.

c) Tritium: The heavy-water reflector system and the containment building.

d) Byproduct: The confines of the location where it is produced (e.g., the core tank, the fission converter tank, the sample irradiation facility or the shielded cell in the secondary chemistry area) and the containment building.

3.1.1.1 Fuel and Cladding

Specifications for the MITR fuel were developed and are maintained by the Idaho National Engineering and Environmental Laboratory (INEEL) for the U.S. Department of Energy. The specification number is TRTR-3, Rev. 2, 1999. This specification in turn incorporates the applicable portions of relevant ASTM (Technical Society Standards), ANSI (American National Standards Institute), ISO (International Standards Organization), MIL-STD (Military Standards), and AWS (American Welding Society) standards as well as drawings prepared by INEEL itself for the manufacture of individual plates. Standard TRTR-3 specifies requirements including the plate loading, void volume, fuel homogeneity, fuel particle location relative to the core of the plate, radiography procedures, cladding and fuel core thickness, evaluation methods, surface finish, fin height, and surface alpha contamination. The specification also covers materials of construction, element assembly from the individual plates, test and inspection requirements, packaging and shipping processes, and acceptance inspections.

3.1.1.2 Primary Coolant System and Core Tank

Table 3-1 lists the design pressures and ratings for the components that comprise the primary coolant system and core tank. In addition, all welds in the system were performed by individuals certified for both the materials involved and the type of weld that was required.

3.1.1.3 Heavy-Water Reflector Coolant System and Tank

The reflector tank is made of 6061 aluminum. The D_2O piping was hydrostatically tested at the time of the reactor modification (1974/1975) as follows: reflector tank and piping to equipment room at 20 psig; main loop pump (DM-1) discharge and heat exchanger (HE-D1) at 50 psig; and balance of piping at 40 psig. These test pressures are at least 150% of the pressures at which the system was designed to operate.

All welding was done by individuals certified for both the materials involved and the type of weld that was required.

3.1.1.4 Fission Converter

The fission converter tank is made of 5083 aluminum. The fission converter design pressure and temperature are below the limits of the ASME Boiler & Pressure Vessel Code. Hence, the fission converter tank is exempt from the specifications of that code. However, ASME code sections II and IX for materials and welding specifications were used to ensure that sound practice was followed in construction of the tank. The fission converter tank and associated piping

Table 3-1

Design Pressures and Ratings for Components of the Primary Coolant System

Component	Design or Test Pressure	Rating
Core Tank 6061 Al	Hydrostatic Test Pressure 60 psig	24 psig at 150° F maximum working pressure
Primary Heat Exchangers (HE-1, HE-1A, HE-1B)	Shell Design - 75 psig at 150° F Tube Design - 75 psig at 150° F Test - 115 psig	Each exchanger has the capacity to cool 1000 gpm of H_2O from 130° F to 115° F when furnished with 815 gpm of cooling water entering at 80° F and leaving at 98.4° F
Primary Piping and Valves 304 stainless steel Transition Sections 6063 Al 6062 Al	Hydrostatic Test Pressure 60 psig Hydrostatic Test according to ASME Unfired Pressure Vessel Code	
Primary Coolant Pumps		1000 gpm each at 108 ft head
Storage Tank 304 Stainless Steel	Design Pressure 40 psig at 300° F	
Auxiliary Pump		10 gpm at 40 ft head
Cleanup System Heat Exchanger		~6 ft^2 of heat transfer area

will be hydrostatically tested at 15 psig. This test pressure is at least 150% of the pressure at which the system will operate.

All welding is done by individuals certified for both the materials involved and the type of weld that is required.

3.1.1.5 Containment Building

The building was designed to withstand 13.8 kPa (2 psig) internal pressure under external temperatures ranging from 43.3° C to -28.9° C (110° F to -20° F). The building was also designed to withstand 0.69 kPa (0.1 psig) of external pressure. The steel used for the shell is ASTM 283 Grade C. The shell, as constructed in 1958, met the American Petroleum Institute specification titled, "Welded Oil Storage Tanks" which is API Standard 12C. It also met Specification API-620, titled, "Recommended Rules for Design and Construction of Large Low-Pressure Storage Tanks." Finally, the building was tested successfully at 2 psig internal pressure. This was done initially by the builder in 1958 and periodically thereafter by the MITR staff.

The design pressure was selected by calculating the energy release of the aluminum-water reaction with 25% of the aluminum in the active section of the fuel elements reacting and then assuming the resulting hydrogen is burned. This energy release gives a pressure rise of 6.9 kPa (1.0 psi) in the building. In addition, maximum likely barometric variations amount to about 6.9 kPa (1.0 psi).

The allowable building leakage rate is 1% of the contained volume (~200,000 ft^3) per day per psig of overpressure.

The containment building is protected against underpressure (vacuum) by two independent sets of vacuum breakers. Each consists of an interior and an exterior spring-loaded breaker in series. The interior breakers are set to open when the internal pressure is between 100 and 250 Pa below atmospheric (-0.015 and -0.036 psig) while the exterior breakers are set to open between 250 and 430 Pa below atmospheric (-0.036 and -0.062 psig). The criterion is to limit the external pressure that could result from a vacuum being drawn within the building to less than the design value of 0.69 kPa (0.1 psig).

Overpressure protection is provided by a pressure relief system that is initiated manually. The criteria are to limit the internal pressure to less than the design value of 13.8 kPa (2 psig) and simultaneously remove radioactive iodine with an efficiency of 95% or greater.

3.1.1.6 Containment Building Ventilation Isolation

The containment building ventilation system is automatically sealed upon detection of abnormal airborne effluent radiation levels. Both the intake and the exhaust ducts are equipped with redundant sets of dampers. These sets are designated as "main" and "auxiliary." Exhaust air is monitored for radioactivity at the entrance to a holdup plenum. The main and auxiliary exhaust dampers are located at the exit of this plenum. The transit delay time in the plenum is such that the main exhaust damper will close in response to a signal from the monitor before the exhaust air that contains the abnormal radioactivity has been discharged. The auxiliary damper closes, also automatically, if the main one fails to do so. The main and auxiliary intake dampers respond to the monitor's signal in similar fashion. Also, the intake dampers are interlocked to close so that effluent can not exit via the intake duct when building ventilation is off.

The criteria for containment building ventilation isolation are that the ventilation dampers close upon detection of abnormal effluent activity and that the closure time is such that the activity is not released.

3.1.1.7 Liquid Waste Discharge Isolation

Liquid waste accumulates in a sump located in the basement of the containment building. The liquid is pumped to one of two storage tanks that are located in a shed in the restricted area adjacent to the containment building. A detector, which is located near the discharge line from the pump, will trip the pump if abnormal liquid effluent radiation levels are detected. This interlock prevents transfer to the storage tanks of radionuclides that emit penetrating radiation.

Other types of nuclides (low-energy beta emitters, for example) would be detected upon sampling the contents of the tank. The tank discharge valve is locked shut except during approved discharge procedures, and tank discharge is also monitored subject to an interlock that can halt the discharge.

The criterion for the liquid waste discharge isolation is that liquid effluent not be released at concentrations in excess of 10 CFR 20 values unless specific exemptions exist in the MITR Technical Specifications.

3.1.1.8 Effluent Path Monitoring

Effluent discharge paths from the MITR are monitored as described above in Sections 3.1.1.6 and 3.1.1.7 of this report. The criteria are that all effluent paths be monitored for radioactivity and that the monitors be interlocked to block the pathway or provide an alarm or another appropriate action so as to preclude releases in excess of 10 CFR 20 values.

3.1.2 Prevention of Core Damage

3.1.2.1 Natural Convection Valves

The MITR is equipped with four natural convection valves that promote the establishment of natural convection cooling upon a loss of forced convection. The criterion is that at least three of these valves open upon loss of forced convection. Diversity exists relative to the natural-convection valves because natural convection paths are also established by the anti-siphon valves. If the two anti-siphon valves are open, natural convection flow is sufficient to remove decay heat.

3.1.2.2 Anti-Siphon Valves

The MITR is equipped with two anti-siphon valves that preclude the siphoning of coolant from the core tank should there be a break in the primary coolant inlet piping. The criterion is that at least one of these valves opens upon loss of forced convection.

3.1.2.3 Emergency Core Coolant System

The MITR is equipped with a manually-initiated emergency core cooling system that sprays water onto the fuel. The criteria are that:

a) The ECCS system be capable of delivering at least 10 gpm within five minutes of receipt of a low level core tank alarm, and

b) Each element position receives at least 20% of the average spray to all elements.

3.1.2.4 Reactivity Coefficients

The MITR is undermoderated and hence has negative reactivity coefficients associated with both the temperature of the primary coolant and the formation of voids in the coolant. The same is true for the heavy-water reflector.

3.1.2.5 Reactivity Insertion Rate Limit

The MITR is designed so that only one control device (one shim blade) can be withdrawn at a time. The criterion for the maximum possible rate of reactivity insertion is that no core damage results from a continuous reactivity insertion at this rate.

3.1.2.6 Maximum Safe Step Reactivity Addition

The criterion is that fuel melting not occur as the result of a power transient created by the maximum safe step reactivity addition.

3.1.2.7 Core Monitoring

The criterion is that it be possible to monitor the reactor power, the core tank level, and the temperature of the coolant in the outlet plenum at all times. Upon loss of offsite power, this could be achieved by use of the emergency electrical power distribution system to supply essential instruments, or the use of battery-powered instruments, or the design of instruments that do not require electrical power.

3.1.3 Reactivity Control and Reactor Safety System

3.1.3.1 Redundancy

The MITR has six shim blades and one regulating rod. The latter is not used for reactor shutdown. The six blades are identical in design and are all located at the same radial distance from the core. Hence, they all have about the same worth. Each has its own drive mechanism and magnet. They can be inserted as a bank. However, only one blade can be withdrawn at a time.

The criterion is that on receipt of a scram signal, all six blade magnets are deenergized and the corresponding blades drop into the core. The shutdown margin criterion (Section 3.1.3.3 of this report) assumes that the most reactive of the six blades fails to insert.

3.1.3.2 Diversity

The MITR may also be shut down by dumping the portion of the heavy-water reflector that is opposite the fuel into a tank that is located below the core. This action would be initiated manually except on loss of electrical power and/or compressed air. In those instances, it would occur automatically.

The criterion is that on receipt of a manual signal, the portion of the reflector that is opposite the core be dumped, thereby inserting sufficient negative reactivity to shut the reactor down.

3.1.3.3 Shutdown Margin

The criterion is that it be possible to shut the MITR down by at least 1% $\Delta K/K$ using shim blades from the cold (10° C), xenon-free critical condition with the most reactive operable blade and the regulating rod fully withdrawn and with all samples in their most reactive state.

3.1.3.4 Reactor Protection System

The reactor protection system consists of both a nuclear and a process safety system. The MITR nuclear safety system consists of six independent channels, three for period and three for power level. The scram logic is such that a short period or a high power signal on any one channel will cause a reactor shutdown. In addition, the channels are monitored for the proper applied voltage to the associated detector. Also, the period channels are monitored for level. If the levels on two out of three of these channels are off-scale (and hence the validity of the period signal questionable), the reactor scrams.

The MITR process safety system monitors primary coolant temperature, level, and flow as well as parameters from the reflector and shield systems. In addition, other components that are important to safety are monitored. The scram logic is such that any one channel will cause a shutdown.

The reactor safety system conforms to the intent of the former IEEE-323-1974.

3.1.3.5 Reactor Protection System Redundancy and Diversity

The electronic design of all six independent nuclear safety system channels is similar. Each channel has its own detector. Cable-runs from these detectors are arranged to preclude a common mode failure of all three period or all three level channels. Redundancy is provided by the use of multiple independent channels of similar design for each function. Diversity is provided by the avoidance of common cable-runs. The MITR's process safety system is also both redundant and diverse. Certain parameters such as primary coolant temperature and flow as well as shield coolant flow are monitored by redundant instruments, any one of which can cause a scram. Diversity is provided by using instruments of differing designs, for example, a capillary sensor and a thermocouple for temperature or an orifice and a pressure device for flow.

The criterion is that there be sufficient redundancy and diversity so that any single failure of any active component will not prevent a safe reactor shut down.

3.1.4 Inspection, Testing, and Maintenance

MITR structures, systems, and components whose integrity is important to the prevention of the release of radioactive material, the prevention of core damage, or to reactivity control are designed to facilitate inspections, testing, and maintenance. Some examples include:

a) Acceptance inspections of fuel elements.

b) Visual inspections of all in-core components for material condition.

c) A pressure test of the containment building to ensure compliance with the allowed leak rate specification.

d) Tests of all interlocks associated with the discharge of radioactive material.

e) Verification of shim blade drop times.

f) Channel checks and calibrations of the nuclear and process safety systems.

Written procedures have been prepared and reviewed for the conduct of all system inspections and tests. Also, approved written procedures are followed for the maintenance of major equipment such as control devices and the ventilation dampers.

Table 3-2 lists surveillance tests, inspections, and calibrations that are related to the structures, systems, and components discussed in this chapter of this report.

3.1.5 Provisions to Avoid or Mitigate Consequences of Fire or Explosion

The MITR containment building and most of the structures therein are built of steel and concrete and/or aluminum and are highly fire resistant. In addition, the following features reduce both the likelihood and consequences of a fire:

a) The reactor is fail-safe and would shut down if fire should damage the reactor protection system.

b) The large volume of water in the core tank would protect the core from a fire.

c) Closed-circuit television can be used to survey the experimental areas from the control room. Also, the control room itself can be monitored by closed-circuit television from outside the containment building.

Table 3-2

Selected Surveillance Tests, Inspections and Calibrations

A. Annual

Emergency Core Cooling System Test

Reactor Building Leak Rate (Biennial)

Charcoal Filter Efficiency Test

Ventilation Damper Inspection

Vacuum Breaker Setpoints

Level and Period Channel Plateaus

Period Channel Calibration

Primary Coolant Flow Calibration

Reactor H_2O Outlet Temperature Calibration

Containment Differential Pressure Calibration

Building Overpressure Scram Test

Core Tank Level Calibration

Reflector Tank Level Calibration

Nuclear Instrument Scram Times

D_2O Reflector Dump Time

Damper Closing Times

Primary Coolant Flow Scram Time

Outlet Temperature Scram Times

B. Quarterly

Effluent Monitor Calibrations

Anti-Siphon Valve Check

Natural Convection Valve Check

Nuclear Instrument Scram Tests

Primary Flow Scram Test

Core Tank Level Scram Test

Reflector Tank Level Scram Test

Core Outlet Temperature Scram Test

Auxiliary Damper Check

Containment ΔP Interlock Test

Dump Valve Test

C. Monthly

Area Monitor Tests

Effluent Monitor Tests

d) Flammable material inventories (paper, wood, solvents, etc.) are minimized for industrial safety. One objective is to identify and remove non-essential combustible material.

e) Detection elements (smoke detectors) alarm in the control room. In addition, pull boxes exist within the building to provide manual notification. The building is equipped with CO_2 extinguishers.

The MITR fire protection program has passive, active, and preventive elements all with the objective of ensuring that safety-related systems can perform their required functions. The program conforms to the intent of ANSI/ANS-15.17-1987.

3.1.6 Quality Standards

Refer to Sections 3.1.1 through 3.1.5 of this report.

3.2 Meteorological Damage

The meteorological history of the MITR site is summarized in Section 2.3 of this report. The principal characteristics were variability and an absence of extreme conditions. The reactor containment building provides more than adequate protection against weather-related phenomena.

3.2.1 Wind Loading

As is noted in Section 2.3.1 of this report, the greatest one-minute sustained wind speed ever recorded in the Boston area was 86 km per hour (54 miles per hour) and the greatest gust was 130 km per hour (81 miles per hour). If the axial profiles associated with these winds are assumed to be constant from the ground to an elevation of 10 meters (32 feet) and if a flat surface is also assumed, then the associated dynamic pressures are 322 Pa (6.74 pounds per square foot) and 726 Pa (15.2 pounds per square foot), respectively. These values are computed using a simplified Bernoulli Equation:

$$\Delta P = \rho V^2/2$$

where ΔP is the pressure change, ρ is density, and V is velocity. This approach yields a conservative answer because:

a) The dynamic pressure is assumed to be the maximum, which it is only at a point on the building (a cylinder) with a tangent normal to the wind direction. The average dynamic pressure would actually be much lower, varying by the square of the sine of the azimuthal angle around the building, assuming inviscid flow,

b) The drag coefficient is assumed to be 1.0, which is conservative for higher Reynolds numbers, such as are involved here,

c) Ground effects, which would reduce the wind speed at lower elevations, are neglected, and

d) Additional support of the building by the concrete shadow shielding is neglected.

Thus, it is apparent that the containment building will not be damaged by winds. The same is true for the ventilation system's exhaust stack which was designed for a 100 mile per hour wind. Other external structures such as the cooling towers are designed to resist such winds. However, they are not essential to safety.

3.2.2 Snow and Ice Loads

As was also noted in Section 2.3.1 of this report, the loading of the 100-year return snowpack is 1.15 kPa (24 pounds per square foot). This is equivalent to 11.7 cm (4.60 inches) of liquid water. The depth of this moisture in solid form would depend on the density of the snow. However, it would be in the range of 3 to 4.5 feet of snow. This amount of moisture, whether in liquid or solid form, presents a danger to structures only if it is allowed to accumulate as could occur, for example, on a flat roof. The torospherical shape of the reactor containment building causes rain to be shed at once and snow to slide off as soon as a small amount accumulates. The presence of any wind accelerates the snow removal process.

3.3 Water Damage

As was discussed on Section 2.4 of this report, the MITR site is not subject to flooding. Moreover, even if water were to accumulate around the containment building exterior, it would not impact reactor safety because the building itself (including the foundation) is water tight.

3.4 Seismic Damage

The seismic characteristics of the MITR site are summarized in Section 2.5 of this report. The major conclusions are:

a) The maximum safe shutdown earthquake for the Boston area has an associated acceleration of 0.15 g. Upon application of an amplification factor for soil type, the appropriate figure for the MITR site is 0.225 g.

b) The maximum earthquake potential is a repeat of the Cape Ann earthquake that occurred in 1755. It had an epicentral intensity of VIII on the MMI scale. This equates to an intensity of VII on the MMI scale in Boston.

c) The soil at the MITR site is not prone to liquefaction.

An analysis of the MITR during a postulated seismic event is given in Section 13.2.8.4 of this report. The major conclusions are:

d) Damage to the core will not occur as long as the core tank remains intact.

e) The tank will remain intact provided that horizontal and vertical accelerations are less than 5.1 g and 3.5 g, respectively. These are well above both those associated with the maximum safe shutdown earthquake and the Cape Ann earthquake. For purpose of comparison, a 5.1 g horizontal acceleration corresponds to an earthquake with a site intensity of XII on the MMI scale. This is well above the Cape Ann event.

f) The ventilation system's exhaust stack should not collapse during a postulated seismic event. However, even if it did, the containment building would not fracture.

g) The shim blades would shut the reactor down during a seismic event.

3.5 Systems and Components

Design criteria for systems and components that are important to safety and which have not been previously discussed in this chapter are provided here.

a) Fuel: The specification for the fuel and cladding is described in Section 3.1.1.1 of this report. It addresses all aspects of the fuel's manufacture. In addition, a limit is observed on the maximum fission density so as to preclude swelling. Parameters enumerated in the specification include:

   (i) Form: UAl alloy or UAl_x cermet with a maximum of 50 w/o uranium in the fuel matrix.

   (ii) Cladding: Aluminum with a nominal clad thickness of not less than 0.015 inches at base of the groove between the fins.

   (iii) Loading: 34.0 + 0.2, -1.0 gram U-235 per plate
   510+3.0, -10.0 grams U-235 per element.

   (iv) Void Volume: 4% to 11%.

   (v) Fission Density: 2.3×10^{21} fissions/cc.

b) Control Device Scram: The time from initiation of a scram signal for a shim blade to go from its full-out position to its 80% inserted position shall be less than 1.0 second. The term "initiation of a scram signal" refers to the time at which the true value of the parameter in question attains its scram setting.

Design bases for other systems and structures important to safety, such as emergency core cooling, containment, and ventilation isolation, are provided in Section 3.1 of this report.

Chapter 4

Reactor Description

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Chapter 4

Reactor Description

4.1 Summary Description

The MIT Research Reactor (MITR) is intended to operate at or below a steady-state thermal power level of 6.0 MW with a primary coolant flow rate of 2000 gpm, a coolant outlet temperature of 55 °C, and a coolant level at overflow. The corresponding limiting safety system settings are 7.4 MW, 1800 gpm, 60 °C, and 4" below overflow. The licensed power is 6.6 MW. The reactor utilizes flat, plate-type, finned, aluminum-clad fuel elements that are highly enriched in U-235. The elements are rhomboid in shape and each contains fifteen plates. The rhomboid shape allows construction of a hexagonal-shaped core that is contained in a tank as opposed to an open pool. Both forced and natural-convection cooling are possible. The former is the normal mode of operation. The latter is used only at power levels of less than 100 kW. Both the coolant and moderator are light water. There are two reflectors. The inner one is heavy water that is contained in a reflector tank that surrounds the light-water core tank. The outer one is made of graphite blocks. The MITR is equipped with a variety of experimental facilities including beam ports, irradiation tubes, and medical irradiation rooms. The beam ports are aligned with radial reentrant thimbles that are welded to the inside of the heavy water reflector tank. This feature maximizes the thermal neutron flux available at the experimental ports. All pneumatic sample irradiation tubes are located in the graphite reflector except one that utilizes a reentrant thimble and terminates in the D_2O reflector. Irradiation facilities can also be installed within the core by replacing a fuel element with an irradiation thimble. These are referred to as in-core sample assemblies or ICSAs. There are two medical irradiation rooms, one located below the core in the basement and one located to the side of the core on the main floor. The latter is driven by a fission converter.

The MITR is not designed to be pulsed.

Table 4-1 lists major reactor parameters. Figure 4-1 is a vertical cross-section of the MITR. As shown in the figure, the reactor core is contained within two concentric tanks and a

Table 4-1

Major Reactor Parameters

1.	Steady-State Operating Power		6.0 MW
2.	Primary Coolant Flow		2000 gpm
3.	Coolant Outlet Temperature		55 °C
4.	Limiting Safety System Settings		
	a)	Power	7.4 MW
	b)	Flow	1800 gpm
	c)	Temperature	60 °C
	d)	Level	4 inches below overflow, which is 10 feet above top of fuel plates.
5.	Coolant and Moderator		Light Water
6.	Reflector		Heavy Water/Graphite
7.	Reactor Type		Tank
8.	Fuel Type		UAl_x cermet
9.	Fuel Geometry		Finned Plate, Rhomboidal Cross-Section
10.	Licensed Steady-State Power		6.6 MW

core shroud. The outer tank, the four foot diameter D_2O reflector tank, is used to maintain a constant D_2O level for neutron reflection. The inner tank contains light water which serves as both the primary coolant and the moderator. The D_2O reflector tank is in turn surrounded by a graphite reflector which extends approximately two feet in the radial direction and is blanketed with CO_2 or an inert cover gas. Numerous experimental ports penetrate through the graphite region to the D_2O reflector tank. Welded into the D_2O reflector tank, and in line with the experimental ports, are the radial reentrant thimbles.

Outside the graphite reflector region is a thermal shield that consists of 1.5 inches of lead supported on either side by two-inch steel plates. The thermal shield is cooled by demineralized water that circulates through cooling coils embedded in the lead. A dense concrete biological shield, approximately 5.5 feet thick, surrounds the thermal shield, and makes the reactor about twenty feet wide overall.

4.2 Reactor Core

Figure 4-2 is a cross-sectional diagram of the reactor core. There are twenty-seven fuel element positions arranged in three rings. There is a central group of three elements, the A-Ring. This is enclosed by a group of nine elements, the B-Ring, which is in turn surrounded by a group of fifteen elements, the C-Ring. Each element is rhomboidal in shape with the result that the core as a whole forms a hexagon. Fuel element positions may be occupied by fuel elements, in-core sample assemblies (ICSAs), or dummy elements. The thimbles and dummies have the same exterior shape as a fuel element. This is done to prevent excessive bypass flow and to limit moderation. There is no limit on the number of non-fueled positions provided that thermal-hydraulic, reactivity, and shutdown margin requirements are met. However, to assure proper flow distribution, all fuel elements positions are occupied when the reactor is operating.

The location of both the control and flux-shaping components are also shown in Figure 4-2. The control elements consist of six boron-impregnated stainless steel shim blades and one cadmium regulating rod. The six blades are used for the attainment of criticality, for the performance of major changes in power level, and for compensation of reactivity changes that occur as the result of xenon and temperature. The regulating rod is used for fine adjustments of power. The regulating rod and all six blades are located on the core perimeter.

The MITR was designed with a provision to install fixed neutron absorbers in the upper half of the core. The objective is to poison the upper half of each fuel element, thereby maximizing the leakage flux into the D_2O reflector beneath the core at the ends of the radial re-entrant thimbles. At present, boron-impregnated stainless steel absorbers are installed in the upper four inches of the radial portion of the core housing so that they are adjacent to the B- and C-Ring elements.

Heat generated by the fission of U-235 is removed from the core by means of the light-water primary cooling system. Coolant enters the reactor through the inlet plenum, flows into the annular region between the core tank and the core shroud, and then moves downward to the bottom of the core tank through the six coolant inlet channels formed by the hexagonal core support housing assembly as shown in Figure 4-2. The coolant is then directed upward through the fuel elements which are held in the core support housing assembly. Water then moves at lower velocity upwards within the space contained by the core shroud to the three exit ports which form the outlet plenum. This plenum is located above the level of the inlet plenum. The coolant passes out of the reactor and flows through two parallel pumps and the primary heat exchangers to form a common line back to the reactor, thereby forming a closed loop. A detailed description of the flow system is given in Chapter 5 of this report.

4.2.1 Reactor Fuel

MITR fuel elements have several unique design features. First, the fuel plates are finned so as to increase the available heat transfer surface and thereby allow their use in a highly compact core. Second, the elements are both radially and axially symmetric so that they can be both rotated and inverted in order to equalize the effect of flux peaks on burnup. Third, the fuel is a cermet that is fabricated with a certain porosity so as to retain gaseous fission products.

Figure 4-3 is a diagram of a MITR fuel element. The MITR utilizes flat, plate-type, finned, aluminum-clad fuel elements that are highly enriched in U-235. Each fuel element consists of fifteen fuel plates. The fuel plates are assembled between two grooved side plates which are 0.188 inches thick and 2.828 inches wide. The fuel plates are 0.080 inches thick with longitudinally-milled fins on both sides to increase the heat transfer surface area. They are 2.552 inches wide, 23.00 inches long and are spaced 0.158 inches apart to form fourteen cooling water passages 0.078 inches wide fin-to-fin. The fins are 0.010 inches high and 0.010 inches wide separated by 0.010-inch wide grooves. Within an element, the spacing between the fuel plates is maintained by the grooves in the side plates. Between elements, end nozzles which are designed to provide structural rigidity for the element maintain the necessary spacing. The fuel elements are rhombic in shape with a perpendicular distance between the flats of 2.375 inches. The overall length, including the end nozzles, is 26.25 inches. Each element has similar nozzles at both ends, thereby allowing it to fit into the lower matrix and to mate with the upper hold-down grid.

[redacted]. The amount of U-235 is evenly distributed among each fuel plate. [redacted] It is clad with 6061 aluminum alloy. The nominal thickness of the clad is 0.015 inches, not including the fin height. Clad thickness may be reduced to not less than 0.008 inches in local areas as an effect of fabrication tolerances and minor manufacturing deviations. The permissible scratch depth is 0.003 inches at the bottom of the groove.

The MITR fuel is made by sintering UAl_x powder. The sintering process results in a final product that is at less than theoretical density. Specifically, the void fraction is between 4% and 7%. This provides space for the accumulation of fission product gases and hence prevents fuel plate swelling. A further advantage is that in the unlikely event of a clad failure, there would not be a sudden release of all fission product gases present in the fuel matrix. Rather, these gases would have to diffuse along the grain boundaries to reach the fuel surface.

The maximum allowed fission density for MITR fuel is currently 1.8×10^{21} fissions/cm^3. However, studies show that this limit is conservative and could be increased to at least 2.3×10^{21} fissions/cm^3 based on the irradiation performance testing of the UAl_x dispersion fuel [4-1, 4-2]. Fuel clad performance associated higher burnup has also been evaluated. Film formation on the fuel clad would present a significant heat transfer resistance because of its relatively small thermal conductivity compared to that of the aluminum fuel clad. Oxidation of the aluminum has been identified as the major contributor to the film growth on the fuel clad. The reaction rate of the oxidation process depends on the thermal (temperature and/or heat flux), hydraulic (flow velocity), and chemical (pH) conditions [4-3, 4-4]. A film thickness of 2 mils or less is recommended which is based on the limit to prevent spallation that could lead to the release of radioactive gas [4-5]. Calculations performed using the MITR operating conditions indicate that the film thickness on the MITR fuel will not exceed 2 mils before the burnup limit of 2.3×10^{21} fissions/cm^3 is reached [4-6].

The MITR fuel elements do not contain any burnable poisons and/or moderating materials. Also there are no partial elements or control rod elements. An instrumented element was fabricated for the initial start-up of the MITR-II in 1974/1975. Thermocouples were crimped between the fins of an otherwise standard element. This element was then used to measure fuel plate temperatures. The thermocouples were subsequently removed and the element was used in the same way as any other element. Another special element was fabricated for the MITR-II's initial startup. This element had removable fuel plates. It was used for flux mapping and peaking factor determination, and is currently in storage at a DOE facility.

Table 4-2 lists materials and physical properties of the MITR fuel.

Operating experience with MITR fuel has been excellent. Thus far, there have been three vendors: Gulf United Nuclear Fuels (43 elements at 445 grams each), Atomics International of Rockwell International (40 elements at 506 grams each), and Babcock & Wilcox (70 elements as

Table 4-2

Material and Physical Properties of MITR Fuel

		Property	Value
A.	Aluminum (6061)		
	1.	Melting Point	660°C
	2.	Softening Point	450°C
	3.	Heat Capacity	915 J/kg °C
	4.	Thermal Conductivity	186 W/m °C
B.	Fuel		
	1.	Mass U-235	[redacted]
	2.	Porosity of Fuel	4-7%
	3	Enrichment	HEU

of April 1999 at 506 grams each). Of the 43 elements made by Gulf United, 42 elements (630 plates) were permanently discharged from the core after peak burnup approached the license limit. The remaining element suffered a clad failure, possibly the result of an inclusion during manufacturing, in 1979. It was removed from the core. All of the AI elements have been permanently discharged except for one which remains in use. Six were identified as showing signs of incipient excess outgassing and two were suspected of this. These eight were removed from service. The remaining 31 elements achieved average burnups of 40-42%. Of the 70 B&W elements introduced to the core through March 1999, one has been removed because of incipient excess outgassing. Twenty-eight others have been discharged upon approaching the fission density limit. Forty-one B&W elements remain in use.

Fresh MITR elements may be stored in the following locations: in the reactor core provided that the reactivity is below the shutdown margin requirement, in the cadmium-lined fuel storage ring attached to the flow shroud, or in the storage safe in the containment building.

Irradiated fuel may be stored in the reactor core provided that the reactivity is below the shutdown margin requirement, in the cadmium-lined fuel storage ring attached to the flow shroud, in the spent fuel storage tank in the basement of the reactor building, in the fuel element transfer cask or other proper shield within the controlled area, or in the fission converter tank.

The principal concerns associated with these storage areas are that criticality considerations be addressed and that the fuel clad not be subject to chemical attack or other forms of deterioration. The criticality issue is addressed by requiring that the k-effective of all storage locations be less than 0.9 [4-7]. The corrosion issue is addressed as follows: elements placed in the storage safe are left in the plastic wrap provided by the manufacturer, an element in the transfer cask is surrounded by air, and the medium in all other storage locations is light water for which the chemical purity is maintained in accordance with the specifications for primary coolant. (Refer to Section 5.2.2.1 of this report.)

MITR fuel is subject to stringent quality assurance. Much of this is performed by the manufacturer in accordance with the requirements of the U.S. Department of Energy, the agency that owns the fuel. In instances where a question exists as to the compliance of a given fuel plate with the design specifications, radiographs are forwarded to MIT for review. Upon receipt at MIT, each element is surveyed for radiation. Table 4-3 lists the surveys that are performed. These are done in accordance with good health physics practice and to satisfy the shipping requirements of the U.S. Department of Transportation. In addition, they serve to confirm the absence of any "tramp" uranium that might be on the surface of an element. Elements are then inspected visually to verify the absence of damage to the clad. This inspection is, of course, limited to accessible surfaces. Irradiated fuel is inspected visually through water for evidence of corrosion and/or clad defects. These examinations are performed quarterly in conjunction with an inspection of all in-core components. One of the tests that is performed is to observe the core with the only source of illumination being Cherenkov radiation. This process can detect foreign material or cladding blisters that might be blocking a fuel element channel.

4.2.2 Control Devices

The MITR uses six coarse-control shim blades and one fine-control regulating rod to adjust reactivity during reactor operation. The location of these control elements is shown in Figure 4-2. Figure 4-4 shows both a shim blade and a regulating rod assembly.

4.2.2.1 Shim Blades

There are six identical shim blades arranged symmetrically with respect to the core. These blades are contained within the light-water core tank and occupy the space between the outermost ring of fuel, the C-Ring, and the core-housing wall. Each blade is worth about 2 beta (1.6% $\Delta K/K$). The blades do not have followers. The full range of travel is such that the tip of the blade is never withdrawn from its guide slot.

The reactor shutdown margin requirement is such that it is possible to shut the reactor down from its most reactive state with the most reactive blade (and also the regulating rod) in the full-out position. In addition, it is acceptable to operate the reactor with only five blades being

Table 4-3

Surveys Performed on Receipt of Unirradiated Fuel Elements

Survey	Location
γ Gross	1 meter from surface of shipping containers Contact with surface of shipping containers
α,β Contamination	Surface and interior of each container
β/γ	1 meter from element Contact with element
α,β Contamination	Surface of element and all accessible plates or portions thereof
α- Survey	Exterior plates of element

functional provided that the remaining one is at or above the average height of the functional ones. Thus, redundancy exists in that the blades are identical and only five are needed both to shut down and to operate. Diversity exists in that each blade can be operated independently of the others. (Note: Diversity is further provided by the capability to dump the heavy water reflector and thereby shut the reactor down. See Section 5.3.1.5 of this report.)

The core configuration can affect the reactivity worth of the shim blades. Specifically, if fresh fuel is placed in the core interior, blade worth decreases. Conversely, if it is placed in the outer rings, blade worth increases [4-8]. However, the magnitude of the change is small, 10-15% at the extreme, and hence there are no limits on core configuration in order to maintain a certain shim blades worth.

The shim blades, which measure 30.625 inches by 7.00 inches by 0.30 inches, are made from 1.1% natural boron impregnated stainless steel. Each blade consists of two 0.125-inch-thick plates separated by spacers so that a 0.050-inch gap exists in the lower 23.375 inches of the blade. This gap, which allows water to circulate between the plates, provides cooling and helps to prevent warping because of radiation damage. The upper end of the blade is attached to a webbed offset plate made of 0.25-inch thick aluminum. These members are welded to a weighted guide rod of low hydraulic cross-section. The weighted guide rod operates in a guide tube bolted to the top flange of the core support housing assembly. An aluminum connecting arm is attached to the guide rod. Affixed to this connecting arm is an armature that is plated with corrosion-resistant material such as nickel or chrome. When the current to the electromagnet on the drive mechanism is cut off, the armature is released thereby dropping the shim blade assembly. Both the weighted guide rod which fits inside the guide tube with a 1/32-inch clearance, and the webbed offset plate which rides in the guide tube slot thereby preventing the guide rod from turning, serve to ensure that the attached boron stainless-steel blade will move freely in its slot without any wear.

The shim blade assemblies move up or down at a constant speed of 4.25 inches/minute. The maximum travel of the shim blades is 21 inches. (Note: For purposes of digital control studies, one shim blade may be driven by a variable speed motor in which case higher speeds are possible. See Section 10.3.2.8 of this report.)

When a shim blade moves in its slot in the core support housing assembly, there must be a means for the water to enter or leave the slot. The required passage is provided by six small through-holes located along the side of each slot that allow the water to flow into holes at the corners of the core support housing assembly. Thus, the hydraulic resistance of the guide tube, not that of the blade slot, acts as a damping force on the shim blade assembly. A small permanent magnet is located at the top of the connecting arm immediately below the main armature. This magnet, which is chrome-plated to prevent corrosion, activates a proximity switch mounted at the top of the guide tube to indicate a "blade-in" condition on the control room indicators. A second proximity switch activates at the height corresponding to 80% blade insertion and is used to measure blade drop times.

Possible degradation mechanisms for MITR shim blades are swelling because of the $^{10}B(n,\alpha)^{6}Li$ reaction, and burnup. Both are discussed in Section 16.3.1.5 of this report. To summarize, swelling has never been observed, and the blades (and the regulating rod) are periodically replaced to recover any loss of reactivity from burnup.

4.2.2.2 Shim Blade Drive Mechanism

The mechanisms for moving the shim blades in or out are driven by electric motors located on top of the upper annular ring. (See Section 4.4.2.2 of this report.) A timing belt sprocket on the drive shaft is affixed by set screws and a copper shear pin. The shear pin prevents mechanical overloading or jamming in the mechanism. Each drive motor power supply is equipped with an individual switch/breaker unit to provide isolation for maintenance.

The drive motor rotates the input shaft of the servo-transmitter and the pinion gear shaft simultaneously. The pinion gear shaft, driven through a chain coupling, penetrates the upper shield ring and the core tank. The pinion gear drives the bevel gear vertical nut drive assembly. As this assembly rotates, the lead screw moves an electromagnet assembly up or down at a rate of

4.25 inches/minute with a maximum movement of 21 inches. By lowering the water level in the core tank approximately two feet, the complete mechanism can be removed.

The electromagnet assembly consists of silicon-insulated wire that is wound and potted into a transit bobbin. This winding is enclosed in an Armco magnet iron housing, capped at both ends by stainless steel and welded leak-tight. The assembly is plated with corrosion-resistant material such as nickel or chrome. The magnet is connected mechanically and electrically to the lead screw. The mechanism is flooded with water to the level of the water seal which is directly below the bevel gear. The upper tube and gear housing are pressurized with helium to ensure that the water remains below the water seal.

The servo-transmitter unit is a small cast-aluminum case that contains two servo-transmitters, a reduction gear, the upper and lower limit microswitches, and the subcritical interlock switches. The servos transmit fine and coarse position indication to the control room.

The following shim blade movements are possible:

a) Individual blades moved outward, one at a time.

b) Individual blades moved inward, one at a time.

c) All blades moved inward at the same time.

d) All blades dropped into the core at once upon de-energizing the electromagnets (scram action).

The MITR has no pulse capability.

4.2.2.3 Regulating Rod

There is one regulating rod. It is located exterior to the core in one of the small circular holes at the corner of the core support housing. The particular hole that is used is located on the fission converter side of the core. The regulating rod's reactivity worth is normally less than 200 mbeta (0.16% $\Delta K/K$). It does not have a follower.

The regulating rod consists of an extension and an absorber section both of which are designed for easy removal and replacement. The extension section is an aluminum pipe 57 inches long with an offset plate and bolting flange welded to the bottom end. The absorber section is

24-7/8 inches long with a 7/8-inch outer diameter. A mating flange and an offset plate are welded at the top. The bottom 18 inches of the absorber section consists of an inner aluminum rod, a cadmium wrap, and an outer aluminum clad.

The range of travel is restricted at the out-limit so that the bottom end of the rod is always contained within the guide hole in the core support housing assembly. The regulating rod's lower limit of travel is approximately six inches above the bottom of the fuel because of its location in one of the holes in the corner of the core support housing assembly. The rod has a maximum range of travel of eighteen inches.

No degradation of the regulating rod has ever been observed as a result of corrosion. Its absorber section is replaced at the same frequency as the shim blades to preclude possible loss of reactivity from burnup.

4.2.2.4 Regulating Rod Drive Mechanism

The mechanism for moving the regulating rod in or out is similar to the shim blade mechanisms. The travel speed of the regulating rod is 4.25 inches/minute. However, for purposes of digital control studies, the regulating rod may be driven by a variable speed motor in which case higher speeds are possible. (See Section 10.3.2.8 of this report.) The regulating rod mechanism differs from the shim blade mechanism in that there is no electromagnet assembly. Instead, it is fixed to its lead screw. Hence, the regulating rod does not drop into the core on receipt of a scram signal. Rather, a scram signal causes it to be driven inwards at its normal speed.

4.2.2.5 Kinetic Behavior of Control Devices

Figures 4-5 through 4-8 show the typical integral and differential reactivity worths of the MITR shim bank (six shim blades) and the regulating rod respectively. The six individual blades have curves very similar in shape to those of the shim bank. However, their magnitudes are a factor of six less, because there are six blades. Also, the curves for individual shim blades are all similar to one another because the blades are identical in design and are all located at the same radial distance from the core center. Table 4-4 summarizes useful parameters relative to these control devices.

The shutdown margin requirement for the MITR is that it be possible to make the reactor subcritical by at least 1% $\Delta K/K$ using shim blades from the cold (10° C), xenon-free critical condition with the most reactive blade and the regulating rod full out and all movable samples in their most reactive state. A sample shutdown margin calculation, which shows that the control device reactivity worths conform to the shutdown margin requirement, is given in Section 4.5.3.3 of this report.

4.2.2.6 Scram Logic and Circuitry

Each of the six shim blades is controlled directly by its corresponding nuclear channel scram amplifier (blade 1 for channel 1, etc.). Each scram amplifier also controls two separate scram relays. One relay cuts off all power to the paired magnet power supply (blade 4 for channel 1, etc.) dropping that shim blade, while the other relay opens the withdraw permit circuit thereby dropping the remaining shim blades. As a secondary action, when the withdraw permit circuit opens, the rundown relays de-energize and drop out. This causes all six shim blade drives and the regulating rod to drive full in. Detailed descriptions of the scram amplifier operation, withdraw permit circuit, and run-down relays are provided in Chapter 7 of this report.

4.2.2.7 Special Features of Control Devices

The MITR shim blade control system is designed to ensure operability and capability to provide safe reactor operation and shutdown under all conditions including that of a single failure or malfunction in the control system itself. This is achieved by using a design that relies on a passive feature (gravity) to achieve the safety function. All six shim blades are coupled to their drives by electromagnets. This coupling provides vertical support. There is no lateral support

Table 4-4

Parameters Associated with Kinetic Behavior of Control Device

	Parameter	Value*
1.	Shim Bank Integral Worth	12.63 beta
2.	Shim Blade Integral Worth	2.26 beta (blade 5)
3.	Regulating Rod Integral Worth	158 mbeta
4.	Maximum Shim Bank Differential Worth	860 mbeta/inch
5.	Maximum Shim Blade Differential Worth	157 mbeta/inch (blade 5)
6.	Maximum Regulating Rod Differential Worth	27.9 mbeta/inch
7.	Maximum Allowed Ramp Reactivity Insertion Rate	3.8 beta/minute (5×10^{-4} $\Delta K/K/s$)

*All values cited assume a blade and rod speed of 4.25 inches per minute. Blade and rod worth figures shown are typical and will vary with fuel loading.

or latch mechanism. Thus, the only action required to effect a safe and rapid shutdown is to de-energize the electromagnets. The shim blades then drop into the core. The system is fail-safe in that:

a) No power source is required to initiate a shutdown.

b) Loss of electrical power automatically results in a shutdown.

c) No mechanical action, such as the release of a latch, is required in order to drop a blade.

d) The blades are always aligned with the blade slot because the range of travel is such that the blade is never withdrawn fully out of the slot.

e) There are six shim blades. Insertion of any five will result in a reactor shutdown under the most reactive core conditions.

The shim blades control system is also designed to preclude other types of failures such as continuous reactivity addition accidents. Protection is as follows:

f) Provisions exist to withdraw only one blade at a time. There is no circuitry that would allow more than one blade to be withdrawn simultaneously.

g) The shim blade (and also the regulating rod) drives are driven by alternating current induction motors. Hence, their speed is coupled to the 60 Hz frequency of the offsite power grid.

h) Both the regulating rod and one shim blade may be used in the conduct of experiments on the digital control of nuclear reactors, particularly spacecraft reactors. For these experiments, the a.c. induction motors are replaced by ones of variable speed. Overspeed protection is then provided both through software and by means of a mechanical speed limiter. (See Section 10.3.2.8 of this report).

4.2.3 Neutron Moderator and Reflector

4.2.3.1 Neutron Moderator

The light water that is used to remove heat from the core also serves to moderate the neutrons. Information on the moderator is therefore given in Section 5.2 of this report which describes the primary coolant system.

An important feature of the MITR's design is that the core is undermoderated. This results in a passive safety feature because an increase in reactor power will raise the temperature of the moderator. This will in turn decrease moderator density and hence reduce the number of neutrons that are thermalized. The result is a decrease in reactivity and a power reduction.

4.2.3.2 Neutron Reflector - Heavy Water

The MITR has two reflectors, an inner one of heavy water and an outer one of graphite. The former is the more important and was installed during the 1974 reactor redesign. The latter was built as part of the original MITR-I reactor that was operated from 1958 to 1974.

High purity heavy water occupies the space between the outer surface of the core tank and the inner surface of the four-foot diameter D_2O tank. The D_2O level in this space corresponds approximately to the upper end of the fuel elements that comprise the core. All surfaces in contact with the D_2O are either stainless steel or aluminum. All free surfaces are blanketed with a helium cover gas. Provisions exist to maintain the purity of the D_2O, to remove heat that is deposited by gamma ray attenuation, and to recombine any D_2 and O_2 gas that is the result of radiolytic decomposition. These systems are described in Section 5.3 of this report.

An important safety feature associated with the heavy water reflector is that it can be rapidly drained to a holding tank located below the core. This action inserts sufficient negative reactivity to shut the reactor down. (Refer to Section 5.3.1.5 of this report.)

4.2.3.3 Neutron Reflector - Graphite

Reactor grade graphite stringers, 3.75 inches wide and either 3.25 or 3.5 inches high, are stacked around the sides of the reactor tank to form a reflector that is 22.25 inches thick (except at the fission converter) and which extends to the radial thermal shield except for a 0.25-inch space that is filled with lead to improve thermal contact. The stack is 7.25 feet high extending about 57 inches below the center-line of the fuel and 30 inches above it.

In the high flux region, one foot above the center-line of the fuel elements to three feet below, the stringers are arranged radially, but are staggered from one course to the next. They are loosely spaced and tapered at the inner ends. In the other regions, the stacking is similar but the ends are not tapered. The stringers are pinned with graphite dowels to prevent lateral or radial movement.

The graphite region is filled with CO_2 or an inert gas at approximately one inch of water above the ambient containment pressure. Provision is made for circulation of this cover gas from the bottom of this region through a condenser that removes trapped moisture and back to the top. Gas seals are made at port liners, the opening for the medical therapy facility water shutter tank, the reflector D_2O inlet and outlet pipes, thermocouple and helium pipes, gasketed flanges of the reflector tank and lower annular ring, and at the aluminum seal plate in front of the fission converter.

Heat generated in the graphite reflector is transferred to both the heavy-water reflector and the thermal shield. The latter is the dominant heat transfer mechanism.

4.2.4 Neutron Startup Source

The MITR has traditionally operated at its licensed power for at least 90 hours per week. More recently, it has operated 24 hours per day, seven days per week with a shutdown of a few days per month for maintenance, instrument tests, refueling, etc. The power history of the MITR is more than sufficient to maintain a strong photoneutron source. Accordingly, there is no need for an installed source and with the exception of the initial startups in 1958 and 1975, none has ever been used.

The MITR's operating license does authorize the possession of two one-Curie PuBe sources and one 150-Curie SbBe source. In addition, MIT's Special Nuclear Material license authorizes possession of additional one-Curie PuBe sources. If there ever were a reactor shutdown of several months and the photoneutron source were no longer viable, the PuBe sources would be used for startup of the reactor. (Note: The SbBe source requires radioactive antimony to function. It is not kept radioactive and hence could not be utilized unless prior arrangements had been made to activate it.) In the event that an installed source were needed, the following procedure would be used:

a) A hollow aluminum tube would be placed in the core center (A-Ring) such that it occupied one fuel element position. This tube would contain perforations so that it would fill with primary coolant and thereby not pose a radiation streaming hazard. Also, the tube would be secured against unintended movement.

b) The source would be loaded into the tube such that it was positioned at or near the core center.

c) Neutron instruments would be checked for operability including the registering of the minimum number of counts necessary for proper use.

d) The MITR would be taken critical.

e) The source would be removed prior to the reactor power being allowed to exceed 500 watts. This is done to preclude melting of the source, which is a possibility at higher reactor power because the PuBe mixture is doubly encapsulated in stainless steel. (Note: Removal of the source will cause an insertion of positive reactivity in the now-critical reactor because water (moderator) fills the space that was occupied by the source.)

4.2.5 Core Support Structure

Figure 4-9 shows the core support housing assembly. It consists of the housing, an upper and lower grid, and the spider. The latter, which is shown in Figure 4-2, is a single piece that consists of an inner hexagonal section and three radial arms. The spider divides the core into four regions. The spider also contains provisions to install fixed absorbers in its upper 12 inches for the purpose of shaping the flux. At present, only the upper four inches of the radial portion of the spider contain absorbers. These suppress a flux peak that would otherwise be created by the coolant in the core outlet plenum.

Figure 4-10 shows the core tank support and the relation between the core support housing, the light-water core tank, and the reflector tank. The top flange of the core support housing assembly is bolted to a support ring which is in turn welded to the inside of the light-water core tank.

The top flange plate of the core housing assembly is used as a platform to secure rigidly the shim blade guide tubes, the hold down grid, the flow shroud, and the grid latch. The top flange plate is penetrated by four check valves that promote natural convection cooling during shutdown or during failure of forced cooling. The valves and their operation are described in Section 6.2 of this report.

4.2.5.1 Design Considerations

The core support housing is designed to ensure that all fuel elements, dummy elements, and in-core experimental facilities are properly secured against all anticipated loads including both the buoyant force of the coolant and the hydraulic forces associated with primary flow. The principal feature for achieving this is a heavy grid structure that is positioned at the top of the core and which serves as a hold-down assembly. This grid is designed to lock down the fuel and other core components during reactor operation and to prevent accidental disassembly of the core by the hydraulic lifting force if a primary coolant pump were to be started during a refueling. The grid structure and its associated interlocks are described in Section 4.2.5.2 of this report.

[redacted] Hence, these elements are held down by gravity with a force of 36.8 N. Upon initiation of primary flow, elements will rise by less than 0.25 inches as a result of the grid. Similar considerations apply to solid aluminum dummies. The forces to which an in-core assembly are subject depend on the assembly's design. These assemblies are normally secured by means of a metal tab that protrudes from the outer thimble and which fits under the grid. A description of the method used to secure a particular assembly is included in the safety evaluation that is written for each such assembly. (See Section 13.2.6 of this report.)

4.2.5.2 Core Component Positioning

Core components are positioned and secured by means of both a lower and an upper grid. The latter is locked in place during reactor operation but can be unlatched to rotate freely when the reactor is shut down to permit refueling operations. However, this rotation is subject to an interlock. Namely, the hold-down grid-plate cannot be rotated unless the shim blades are fully inserted because rotation of the hold-down grid-plate mechanically locks the shims in this position. While the hold-down grid-plate prevents large motions of the absorber spider, the hold-down grid-plate latching mechanism holds down and prevents small motions of the absorber spider. This grid-plate latching mechanism is mechanically operated from outside the light-water core tank, where it is electrically interlocked with both the withdraw-permit circuit and the primary coolant pumps. The grid and these interlocks provide the following safety features:

a) Whenever the hold-down grid is unlatched, a reactor scram signal is present.

b) Whenever the hold down grid is unlatched, the primary coolant pumps cannot be operated. (If the pumps were running, the reactor scrams and the pumps trip.)

c) The grid cannot be physically rotated unless all six shim blades are inserted. This ensures that the blades are fully inserted before a refueling is initiated and that they remain inserted during refueling.

d) The opening in the grid is such that at most three (and usually only one) fuel element positions are accessible. Therefore, three elements is the maximum number that could be accidentally discharged were primary flow to be somehow initiated while the grid was unlatched and rotated. This is less than the minimum number (four) that is allowed outside of an approved storage area, and hence they could not in themselves present a criticality hazard.

In summary, the hold-down grid and its associated interlocks require that the MITR be shut down with all shim blades inserted and that the primary coolant pumps off before a refueling can be initiated. Conversely, all fuel elements and other core components must be secured in the core before the grid interlock can be cleared to allow the establishment of primary flow and control device withdrawal.

4.2.5.3 Materials of Construction

The core support housing is made of aluminum-6061, the same material from which the core tank is made. Radiation damage effects are discussed in Section 16.3.1.4 of this report.

4.2.5.4 Design Features

The core support structure incorporates a number of design features that accommodate other systems. These are:

a) As noted previously, the top flange plate of the core housing assembly serves as a platform for the shim blade guide tubes, the hold-down grid and its latch, the natural circulation valves, and the flow shroud. (See Sections 4.2.2.1, 4.2.5.2, 6.2, and 4.3.1.6 for a description of each of these items.)

b) The spaces between the six outer surfaces of the hexagonal-shaped core support housing and the light-water core tank form six coolant inlet channels for the primary coolant as it moves from the inlet plenum to the bottom of the core.

c) Six holes, one at each corner of the support housing, provide a passage through which water in the shim blade slots can be ejected when a blade is inserted. One of these holes is utilized for the regulating rod. The others are normally empty, except for coolant. However, small diameter fission detectors can be placed in them without altering blade drop times. These detectors have been used for flux mapping.

4.2.5.5 Movable Core Support

The MITR core is not designed to be moved.

4.3 Reactor Tanks

The reactor core, the light-water coolant/moderator, the heavy water reflector and its helium cover gas, and the control devices are all contained within two concentric tanks. These are the light-water core tank and the heavy-water reflector tank. Figure 4-1 shows the arrangement of these tanks. The light-water tank is supported by the heavy-water one. It in turn rests on the top flange of the lower annular ring, which is a section of the reactor's biological shield. (See Section 4.4.2.2 of this report.) The lower annular ring rests on the inner section of the radial thermal shield (1.5 inches of lead surrounded on either side by two-inch steel plates). The tanks are attached to the lower annular ring by 23 one-inch studs. The lower annular ring, in turn, has a flange that is bolted to the inside steel cylinder of the radial thermal shield.

All penetrations through the light-water tank are above the core. Also, those that are capable of creating a siphon are equipped with a siphon-breaker as described in Section 6.3 of this report. The result is an exceptionally safe design. Both tanks are, as described below, designed against failure that could cause loss of integrity. Both tanks would have to fail in order for the core to be uncovered. A pipe break or a single tank failure could not result in such a scenario.

4.3.1 Light-Water Core Tank

The light-water core tank was manufactured and installed in 1974. The MITR's core shroud and flow guide, which are attached to the core tank and the core support housing respectively, are also described here.

4.3.1.1 Design Considerations

The design pressure of the light-water core tank is 60 psig. The core tank is rated for 24 psig at 150 °F. The maximum hydrostatic pressure, which occurs at the bottom of the tank, is 5.4 psig. The maximum hydrodynamic pressure, which occurs with a primary coolant flow of 2100 gpm, is 14.2 psig. Thus, the design pressure is adequate by a factor of 1.7.

The pH of the primary coolant is slightly acidic, typically 5.5 - 6.5. This minimizes corrosion of aluminum. Radiation damage to the core tank is discussed in Section 16.3.1.6 of this report. Correlations developed at the Brookhaven National Laboratory [4-9] have been applied to the MITR's operating history. The material properties of the present light-water core tank will remain satisfactory for at least twenty more years with the reactor operated at 10 MW [4-10].

4.3.1.2 Shielding/Adequacy of Depth

The distance from the top of the MITR fuel plates to the overflow pipe is 10.3 feet. Shielding of the reactor core is normally provided by the coolant volume that is above the core and the reactor top shield lid which is described in Section 4.4.2.2 of this report. The lid is required to be in place if the power level is to exceed 100 kW. There are two scenarios where the coolant volume alone is used for shielding. These are:

a) Refuelings/In-Core Maintenance:

The reactor is shut down and sufficient time has elapsed for short-lived radionuclides that are present in the coolant itself to have decayed. In this case, the radiation level on the coolant surface will be only a few mrem/hour. (Note: The short-lived nuclide of concern is Na-24 which is produced by an (n,α) reaction off Al-27 and which is soluble in the coolant.)

b) Natural Convection Mode:

The reactor is operating at a power level of less than 100 kW. In this case, the radiation level on the coolant surface would be 1.3 R/hr.

If the reactor is shut down, but Na-24 activity is still significant, then administrative procedures established by the MIT Reactor Radiation Protection Office are observed if it is necessary to work above the core. These procedures might establish stay times or restrict removal of shielding to a single port plug that penetrates the top shield lid. (See Section 4.4.2.2 of this report.) If the reactor is to be operated at power levels in excess of 100 kW, then the reactor top shield lid, which is a lead-filled weldment, must be in place.

The depth of the coolant is sufficient to provide the necessary pressure for coolant pump operation. The pumps are located in the reactor basement at a level below the core tank. The net positive suction head is given by the relation:

$$NPSH = P_{system} + P_{gravity} - P_{friction} - P_{sat} \tag{4-1}$$

where

P_{system} is the pressure on the coolant surface;

$P_{gravity}$ is the static gravity head;

$P_{friction}$ is the friction head loss in the suction piping and the pump inlet; and

P_{sat} is the vapor pressure of the coolant.

For coolant at 55 °C, and 2000 gpm, the values of the four parameters on the right side of the above equation are 1.013×10^5 Pa (1 atm), 6.937×10^4 Pa, 3.782×10^4 Pa, and 1.576×10^4 Pa. Thus, the NPSH is 1.171×10^5 Pa or 17.0 psi.

4.3.1.3 Design

The core tank is constructed of high purity aluminum (6061 alloy) and is fabricated in two sections to accommodate the polar crane's lifting limit. The bottom section of the tank has two diameter-transitions as shown in Figure 4-1. The bottom portion has an inside diameter of 20 inches with a 0.25-inch wall thickness. Near the top of the core, it widens to an inside diameter of 45 inches with a wall thickness of one inch. An elevated support ring, which supports the core support housing assembly and the core shroud, is welded to the bottom of the 45-inch diameter section. Immediately below the top flange of the core tank is the eight-inch primary coolant inlet pipe that leads to the inlet plenum. This plenum is designed to form a fan-shaped water flow so that the coolant will flow around and downward in the annular space between the core tank and the core shroud. The overall height of the core tank bottom section from the dished bottom to the top flange is 102.375 inches. The tank's design pressure is 60 psig.

The top section of the core tank has an inside diameter of 44 inches with a wall thickness of 0.75 inches and overall height of 50 inches. Its diameter matches that of the top flange of the bottom section of the tank. Immediately above the bottom flange of the top section is the outlet plenum and its eight-inch return pipe. The water level in the core tank is controlled by a two-inch standpipe or overflow pipe which maintains the level of coolant at three inches below the top of the tank. Immediately above the level of the overflow pipe is the off-gas pipe, a two-inch radially perforated pipe that is used to remove N-16 and Ar-41 gases from above the reactor pool. Surrounding the tank, 19 inches below the top lip, are six penetrations, sixty degrees apart, for the shim blade drive mechanisms and one penetration for the regulating rod drive. The regulating rod is on the center-line of the fission converter. Penetrations also exist for the tank level probe and pressure taps.

4.3.1.4 Location of Penetrations

The locations of all penetrations are given in the preceding section of this report. They are all located well above the core and those that are capable of creating a siphon are equipped with siphon breakers.

4.3.1.5 Assessment of Radiation Damage

Radiation damage to the core tank is a function of the fluence which in turn depends on the power history. As noted in Section 4.3.1.1 of this report, an assessment has already been made of the radiation damage that will occur during the next twenty years of MITR operation [4-10]. Material properties of the core tank will remain satisfactory.

A failure of the light-water core tank would not result in uncontrolled leakage of contaminated primary coolant. Any leakage would be into the heavy-water reflector. Also, a primary system pipe break at a location exterior to the core tank would result in a limited spill of coolant. The extent of the spill would be limited because the coolant pipe penetrations are near the top of the tank. There are no floor drains so all spills are cleaned up locally.

4.3.1.6 Core Shroud

The core shroud, which is shown in Figure 4-1, serves to separate the inlet and outlet primary coolant flow paths. It is bolted to the core tank support ring which is welded to the inside of the core tank. The core shroud has an inside diameter of 41 inches with a wall thickness of 0.75 inches and an overall height of 75 inches. It is constructed of 6061 aluminum alloy. The top section of the core shroud is sealed to the core tank by three rings of polyethylene packing. This packing is compressed to form a tight seal by the packing retainer ring bolted around the top edge of the core shroud.

There are two float valves installed at the top of the core shroud. These are the anti-siphon valves that, in the event of a pipe break on the inlet side of the primary piping, will prevent coolant from being drawn back through the annular space thereby uncovering the reactor core. If such an accident occurred, as the water reaches the level of the anti-siphon valves, air replaces the water exiting the inlet plenum. Otherwise, water would be siphoned from the inlet plenum through the inlet pipe to the break point.

A fuel storage ring is bolted to the bottom flange of the core shroud. This liner provides twenty-nine rhombic-shaped fuel element storage positions around its periphery. Each storage position is constructed of 0.020-inch cadmium that is clad in 0.063-inch thick aluminum and welded to the tank liner wall. The bottom of each position is fitted with an aluminum-cadmium spacer.

4.3.1.7 Flow Guide

The initial low-power testing of the MITR-II in 1974 revealed that small power fluctuations occurred when two primary coolant pumps were operating. The cause of these fluctuations was determined to be lateral movement (flutter) of control blades caused by the water turbulence of the coolant as it exited the fuel elements. A flow guide was added at the top of the reactor core in order to prevent this flutter.

The flow guide is shown in Figure 4-1. It is 30-inches high and hexagonally-shaped with a perpendicular distance between the flats of 15.25 inches. It is attached to the retaining ring by twelve studs and nuts. The entire assembly is constructed of 6061 aluminum.

4.3.2 Heavy-Water Reflector Tank

The heavy-water reflector tank was manufactured and installed in 1974.

4.3.2.1 Design Considerations

The design pressure of the heavy-water reflector tank is 40 psig. The material condition of the tank is excellent as described in Section 16.3.1.7 of this report.

4.3.2.2 Shielding/Adequacy of Depth

The heavy-water does not provide shielding to personnel working above the core. This is evident from the geometry of the core as shown in Figure 4-1. However, the attenuation provided by the heavy water that is opposite the core is an important, though not crucial, factor in the limitation of radiation on the reactor floor exterior to the biological shield. The following factors are relevant:

a) The dump valve is interlocked with the reactor withdraw permit circuit so that whenever the dump valve is open (and hence the reflector is dumped and the vacated space filled with helium), the reactor is in a shutdown condition. Hence, the potential loss of the attenuation provided by the heavy water is not an issue during reactor operation.

b) Measurements made after a simultaneous scram of the reactor and a dump of the heavy-water reflector show an increase in radiation levels on the surface of the biological shield by a factor of about two. This is not significant because the reactor is shut down and hence radiation levels are already low. (Note: The increase may be higher, a factor of ten, in the vicinity of beam ports that are aligned with reentrant thimbles. These locations are not accessible except with supervision by the Reactor Radiation Protection Office.)

The bottom section of the heavy-water reflector tank is penetrated by an eight-inch pipe for filling and dumping a portion of the reflector and a 0.75-inch pipe for draining the D_2O shutter. There is also a 3-inch pipe for D_2O circulation. The D_2O shutter is one of four shutters that control entry of a neutron beam to the medical therapy facility located below the MITR. Fill and void operations of this shutter are controlled at the medical facility control panel. Figure 4-1 shows one of the re-entrant thimbles that are welded to the inside of the reflector tank so as to line up and mate with ports that penetrate through the biological shield and graphite reflector to the outside edge of the reflector tank.

The tank also contains a helium cover gas blanket. The helium cover gas is cycled through a recombiner to control the concentration of deuterium. The helium is circulated via four one-inch penetrations through the core tank bottom section flange. A constant-pressure gasholder, at approximately one inch of water above atmospheric pressure, serves as a surge and storage tank for the blanket system.

4.3.2.3 Location of Penetrations

Most penetrations for the heavy-water reflector tank are located below the tank. This is necessary in order to make it possible to shut the MITR down by dumping the heavy water into a holding tank located below the reflector.

4.3.2.4 Assessment of Radiation Damage

Radiation damage to the heavy-water reflector tank is within the envelope of that to the light-water core tank because the latter is exposed to a higher neutron flux, especially fast neutron flux. Accordingly, the material properties of the heavy-water reflector tank will remain satisfactory over the projected life of the MITR.

Failure of the heavy-water reflector tank would not result in a significant radiation hazard except for tritium. Any spilled heavy water would be contained within the reactor equipment room because there are no flow drains and the sump is surrounded by a dam. The only path whereby radioactivity could be released to the environment is by evaporation and transport through the containment exhaust ventilation. This scenario is discussed in Section 13.2.9 of this report.

4.4 Thermal and Biological Shield

4.4.1 Design Basis

The MITR's thermal and biological shields were constructed in 1956-1958 as part of the original reactor. The former is made of steel and lead, the latter of high density concrete. The design basis was 0.2 mrem/hr on the shield surface at the initial reactor operating power of 1 MW. At the MITR-I final operating power of 5 MW, it was 1.0 mrem/hr.

Neither the thermal/biological shield nor the reactor power level were changed during the modification that occurred in 1974 - 1975. However, two changes were made that altered the effectiveness of the shields. These were:

a) The core was made more compact and the heavy water reflector was installed. These changes had the effect of reducing general area radiation levels at the surface of the shield.

b) The re-entrant thimbles were installed thus increasing the flux available to experimenters at the beam ports. This had the effect increasing local radiation levels at some of the beam ports.

The net effect of the 1974 - 1975 modification was that the existing biological shield continued to fulfill its original function which was to permit twenty-four hour per day access to the reactor floor and basement areas. However, it was necessary to augment the shield with concrete slabs and/or lead bricks in the immediate vicinity of some of the beam ports.

For 6.0 MW operation, the surface dose will be 1.2 mrem/hour. This permits unrestricted access to those using or operating the facility. For a 40 hour week and 50 weeks/year, the dose received by someone continuously present at the shield surface would be 2400 mrem. Actual exposures are much less because no one works in such proximity to the shield surface for such durations. Installation of the fission converter facility will improve shielding near adjacent beam ports. It is expected that local radiation levels will decrease at some locations.

To summarize, the radiation source term is the reactor core, spent fuel in the fuel storage ring, and activated core components. There are no other major radiation sources within the core tank. The shields are made of solid materials, mostly steal, lead, and high density concrete. Water-filled tanks are not used as part of the shields although such tanks are employed for on/off control of some beams. The shields were designed for operation of the original (MITR-I) reactor at its ultimate power level of 5.0 MW. However, these shields remain sufficient for reactor operation at 6.0 MW.

4.4.2 Description

The MITR has both a thermal and a biological shield. The former consists of a radial cylindrical section, a bottom section, and a fission converter section. (<u>Note</u>: The radial cylindrical section is the portion described briefly in Section 4.1 of this report.)

The biological shield consists of several feet of dense concrete which is made of Portland cement, magnetite iron ore, and steel punchings with the mixture having a density of approximately 280 lb/ft^3. The section surrounding the reactor tank and radial reflector includes the lower and upper annular rings, the upper shield ring, upper shield access ring, and the top shield lid. These are shown in Figure 4-10.

Descriptions are given below of the separate components of both shields.

4.4.2.1 Thermal Shield

The radial section of the thermal shield consists of an inner steel cylinder that has an inside diameter of eight feet and is two inches thick, a lead-filled space which is 1.5 inches thick and in which two cooling coils (one being a spare) are embedded, and an outer steel cylinder which is also two inches thick. Openings are provided for beam ports and part of the side is cut out to form an opening to the fission converter. All inside surfaces that are exposed to the neutron flux are lined with 0.25 inches of boral. The openings for the beam ports are lined with cadmium and have cadmium collars that overlap the boral. The top of the inner steel cylinder supports the lower annular ring which in turn supports the two concentric reactor tanks.

The bottom of the thermal shield is similar to the radial section in that it also consists of two inches of steel, 1.5 inches of lead with two embedded cooling coils (one being a spare), and another two inches of steel. It connects to the radial thermal shield so as to form a container for the graphite reflector as well as for the light and heavy water tanks. Penetrations are provided for the D_2O reflector inlet and outlet pipes and for the basement medical facility shutter tank.

The opening for the fission converter in the radial thermal shield has a frame to which is bolted a 1/16 inch aluminum sheet with a gasket. This forms a gas-tight seal between the graphite reflector region and the fission converter. This sheet has a 14-inch by 14-inch by 14-inch gas-filled re-entrant thimble that allows access from the fission converter to within approximately one inch of the reflector tank wall.

A cadmium/boron shutter is located in a guide-support directly in front of the fission converter. This shutter is opened and closed by a motor-driven mechanism located near the top of the reactor. Use of the fission converter is controlled by this shutter.

4.4.2.2 Biological Shield

The lower annular ring is a steel weldment filled with dense concrete which has a minimum inside diameter of 56 inches, an outside diameter of approximately eight feet, and a height of four feet. The lower annular ring surrounds the reactor on all but the fission converter side of the facility. The lower surfaces are covered with 0.25 inches of boral. The lower annular ring has a 4-inch thick, water-cooled, lead thermal shield which lines the inner surface over the lower eighteen inches. Sleeves are provided for six vertical thimbles which slant downward through it into the graphite reflector. There are also conduits for thermocouple leads and helium gas pipes.

The upper annular ring is a steel weldment filled with dense concrete with an inside diameter of 58 inches and an outside diameter in excess of eight feet except on the fission converter side where a segment is omitted. Its average height is 18 inches with a maximum height of 32 inches. The flat side facing the fission converter is approximately 19 inches high in order to accommodate the eight-inch light water inlet and outlet coolant pipes. Plugged holes are provided for access to the vertical thimbles which slant through the lower annular ring and into the graphite region. The upper ring is supported by several steel blocks bolted to the lower annular ring flanges.

The upper shield ring is a steel weldment filled with dense concrete and lead with an inside diameter of 46 inches and a height of 41 inches. The outside diameter is stepped to prevent radiation streaming. The upper shield ring rests on top of the upper annular ring. It has seven radial penetrations to accommodate the shim blade and regulating rod drives. The smallest outside diameter of this stepped structure is 58 inches while the largest is 68 inches. The bottom contains four inches of lead for biological shielding.

The upper shield access ring is a steel weldment filled with lead which is 68 inches in diameter and 16 inches high. It has an inside diameter of 44 inches. The ring has numerous radial access ports to facilitate manipulations inside the core tank. Around the top surface of the upper shield access ring is a machined groove 3 inches wide and 2.375 inches deep to accommodate a captive ball bearing ring. The top shield lid rotates and rests upon this ball bearing ring.

The top shield lid is a steel weldment filled with 10.38 inches of lead. There is one 16-inch, one 12 inch, and two 8-inch access holes which are stepped. Within the 16-inch access hole is a smaller access hole 6 inches in diameter. In addition, there are three lifting eyes 120 degrees apart. The lid is 45.6 inches in diameter at the bottom and then widens to 66.4 inches. The weight of the top shield lid is approximately 5.5 tons.

Biological shielding on the sides of the reactor consists of dense concrete about 5.5 feet thick except where displaced by beam holes, port boxes, shutters, utility pipes or the fission converter. Some concrete is also displaced by twenty-two vertical 4.875 inch holes which are accessible from the reactor top and are used for dry storage.

The reactor faces are sheathed with 0.375 inch steel plate, with removable panels that provide access to the utility shelf, port utility boxes, and storage holes.

There are 50.5 inches of dense concrete between the bottom of the thermal shield and the ceiling of the medical therapy room. The only exceptions to this are the water shutter tank and the D_2O and H_2O pipe penetrations. The small enclosure where the pipes turn, and the ducts to the equipment room from this enclosure are filled with removable shielding.

4.4.3 Materials

The reactor core is surrounded by both a thermal and a biological shield. The former is made of steel and lead. The latter is constructed of high-density concrete which consists of

Portland cement, magnetite iron ore, and steel punchings. The average density of the mixture is 280 lb/ft^3. Information on the nuclear and shielding properties of these materials is available in standard references such as that by Shultis and Faw [4-11].

Both shields have been in place since the MITR's initial criticality in 1958. Neither has incurred any damage from radiation, heating, or material disassociation. Moreover, with the exception of heating effects on lead, no damage of these types would be expected for the materials utilized in the MITR shields. Lead has both a low softening and a low melting point. Hence, the portion of the thermal shield that is made of lead could be deformed if the heat deposited through radiation attenuation were not removed. This is the reason for the installation of redundant sets of cooling coils within the lead portion of the thermal shield. These coils are part of the shield coolant system which is described in Section 5.4 of this report. Other relevant information on the MITR shields is as follows:

a) The thermal shield and the inner portion of the biological shield are subject to a neutron flux and hence some induced activity would be expected. However, the activity is not in accessible locations and hence does not constitute a hazard to personnel. Also, boron and cadmium linings minimize the induced activity.

b) Penetrations through the shields for instrument lines or coolant flow are designed to preclude radiation streaming. Penetrations are stepped or, where that is not possible, slanted.

c) Beam ports for experiments are the exception to the design guideline that all penetrations through the shield be designed to avoid radiation streaming. Of necessity, the ports are straight paths through the shield. Hence, experimental ports are provided with some further protection such as water-filled shutters, beam stops, or temporary shielding.

d) Fuel that will be reused in the MITR core is stored in the fuel storage ring which, as described in Section 4.3.1.6 of this report, is contained within the light-water core tank. Hence, it is shielded in the same manner as the reactor core.

e) Fuel that is spent and is awaiting return to the U.S. Department of Energy is initially placed in the fuel storage ring, and once decay heat levels have declined, transferred to the spent fuel storage pool, a 21-foot-deep steel-lined concrete tank that is located in the reactor basement. The fuel is stored in cadmium-lined racks at the bottom of this pool. Hence it is shielded by about 20 feet of light water.

4.4.4 Prevention of Ground Activation

The MITR is designed to preclude the possibility of either soil or ground water activation. Specific features that accomplish this are as follows:

a) The reactor core is located at the center of the gas-tight, steel-shell containment building. As described in Section 4.1 of this report, the core is surrounded by two concentric tanks, the thermal shield, and the biological shield. The space exterior to the biological shield in the radial direction is for experimenters. That below the biological shield is for the medical therapy room that is located in the reactor basement. Thus, there is no contact or even proximity between the systems that are radioactive (core, primary, heavy-water reflector) and either soil or ground water. Any leakage would have to go from the biological shield through the open experimental areas (where it would be visible) and then through the containment building itself (which is leak-tight).

b) There are no underground waste tanks associated with the reactor. Also, there are no underground ducts. (Note: Exhaust ventilation is ducted from the equipment room in the basement of the containment building to the base of the stack. However, the duct is entirely contained in an accessible concrete structure. The duct is not in contact with soil.)

c) The secondary coolant pipes do run underground. However, secondary coolant is normally not radioactive and it is monitored for radioactivity as described in Section 7.7.2.2 of this report.

d) The spent fuel storage pool, which is described in Section 9.2.2.1 of this report, does penetrate into the soil below the containment building. It is a 21-foot-deep, steel-lined concrete tank that is located in the containment building's basement. The steel liner is 3/4 inches thick, and the bottom of the tank is 2.0 inches thick and also made of steel. There has never been any leakage from this tank. Moreover, the water table is such that, were a leak to develop, the leakage would go into the tank, not out. Water in the spent fuel pool is maintained at the same standards of chemical purity (pH, conductivity) as the primary coolant. The concentration of radionuclides in this water is normally less than the minimum detectable. (Components that have been immersed in the heavy water reflector and which may have absorbed tritium are not placed in this tank.) Soil samples were taken of the reactor site in 1956-1958 prior to the start of operation of the MITR. These remain in sealed canisters. Soil samples taken from the site in 1999 revealed no detectable activity.

4.5 Nuclear Design

Many research reactors function by defining a limiting core configuration as that core which yields the highest power density for the specified fuel. Other configurations are then allowed provided that they are within the envelope of this limiting one. For the MITR, the limiting core would be the one with the maximum number of non-fueled positions because the power density in the fuel elements would be at its highest. Criticality considerations restrict MITR operation to a maximum of five non-fueled positions. Such a core was installed and operated following the initial startup of the MITR-II. A more routine configuration is three non-fueled positions. However, cores have also been installed that had either two or four non-fueled positions.

The power peaking factors used for the thermal-hydraulic analysis given in Section 4.6 of this report are from the core with five non-fueled positions and hence are worst case. Limiting thermal-hydraulic conditions (e.g., equations 4-26 and 4-37 for forced convection flow) are then developed as a function of the maximum power deposited in a fuel channel. Cores other than the limiting one are evaluated by verifying that these limiting core conditions are met. An analysis is then done of every fuel channel in a proposed core to be certain that these limiting conditions are not exceeded. If so, the proposed core is acceptable. The principal steps in the analysis are:

a) A three-dimensional (r-θ-z geometry) model of the core is developed. The model specifies the physical layout of the core components (fuel, control devices, moderator, housing, reflectors, etc.), the material composition of each component (uranium, aluminum, water, boron, etc.), and the number density of each composition. Provisions exist to include xenon and other fission products.

b) A numerical code is used to obtain the K-effective, flux distribution (three energy groups), and power density distribution of the modeled core. The codes used for this purpose are ones obtained from the Radiation Safety Information Computational Center (RSICC at Oak Ridge). CITATION, which is based on diffusion theory, is currently in use at the MITR [4-8]. MCNP, a Monte Carlo code, is being evaluated [4-13].

c) The calculated power distribution is modified by the application of shape correction factors. These factors are needed because a diffusion theory code such as CITATION may not predict the neutron flux (and hence the power density) correctly at material interfaces [4-8]. The calculated factors are obtained by comparing a calculated power density with a measured one. Correction factors are essentially unity in the core interior where the calculation is expected to be accurate because there are no abrupt changes of materials. The factors depart from unity for the outer fuel plates of the C-ring because they are in close proximity to the control devices and the reflector. Similarly, these factors depart from unity at the top and bottom of the core because of the inlet and outlet plenums. The correction factors, which are a ratio of measured to calculated power density, are dimensionless. They are assumed to remain constant and can be applied to cores other than the one for which they were derived [4-8]. (Note: Initial evaluation of MCNP outputs indicates that shape correction factors will not be required. This is to be expected because MCNP does not rely on the diffusion theory approximation.)

d) The proposed core's thermal-hydraulic limits are evaluated using the corrected power distribution for every fuel channel. If all channels pass, then the proposed core is acceptable.

The above approach is computationally intensive. However, it offers flexibility in terms of both in-core experiments and fuel utilization.

4.5.1 Normal Operating Conditions

4.5.1.1 Core Components

Core components are described in Section 4.2 of this report.

4.5.1.2 Planned Core Configurations

There is no set of pre-planned core configurations for the MITR. Rather, any configuration is acceptable provided that certain criteria are met. These include:

a) Certain thermal-hydraulic parameters must not be exceeded for any channel in the core.

b) Each of the twenty-seven positions within the core must contain a fuel element, or a solid aluminum dummy, or an approved in-core sample assembly (ICSA). (Note: Also acceptable is a neutron source tube as discussed in Section 4.2.4 of this report.)

c) The shutdown margin requirement is met.

d) No fuel element, or any portion thereof, exceeds the fission density limit.

4.5.1.3 Reactor Operating Characteristics

MITR fuel management policy is, in general, to place fresh fuel in the B-ring where peaking factors are lowest and to place partially-spent fuel in the A- and C-rings where peaking tends to be greater. (Refer to Figure 4-2.) The net result is that power is concentrated in the core interior when a core is newly installed. As the fuel depletes, the average height of the shim bank increases. All six shim blades are located in the core periphery. Hence, as the bank rises, power shifts from the A- and B-rings to the C-ring. Detailed analysis of the power distributions of the selected cores are available [4-8]. These studies have shown that as shim blades and/or fixed absorbers are raised, the following occurs:

a) The axial spatial location of the point of maximum flux will be higher.

b) The magnitude of this maximum will be decreased.

c) The magnitude of the thermal neutron flux available at the beam port reentrant thimble tips decreases.

d) Power density decreases in the A- and B-rings but rises in the C-ring. The average power density for the core decreases as the same total power is being produced in what is, effectively, a larger volume.

These same studies have established that insertion of fresh fuel in a given position increases the power density in that position at the expense of neighboring positions, and that replacement of a solid dummy and/or experimental facility with fuel increases the power density in the neighboring elements.

4.5.1.4 Effect of Fuel Burnup

The MITR operates 24 hours per day, seven days per week. Hence, refuelings are frequent. These may involve any or all of the following: replacement of spent fuel with fresh or partially used fuel, shuffling of fuel to achieve better overall burnup, rotation of individual elements to offset the effect of radial flux gradients, and inversions (flipping) of individual elements to negate the effect of axial flux gradients. Detailed analysis of these refueling strategies are available [4-8].

The effect of fuel depletion on reactivity has been quantified [4-8]. The change in reactivity with core energy production is -0.25 mbeta/MWH. Plutonium buildup is not of significance because of the level of the fuel's enrichment.

4.5.1.5 Kinetic Behavior/Requirements and Features of Control Devices

Information on the integral and differential reactivity worths of the MITR's shim blades and regulating rod is given in Section 4.2.2.5 of this report. The integral worth of the shim bank is typically 12 beta. (Note: As discussed in Section 4.2.2.1 of this report, the worth of the shim bank will vary with the radial power distribution. The typical range is usually 11.5 - 13.0 beta.) Figures 4-5 and 4-6 showed the shape of the integral and differential shim bank reactivity worth curves. These shapes are a function of the energy and magnitude of the flux to which the blades are exposed. The high differential worth that exists in the center of travel corresponds to a maximum in axial flux shape. Figures 4-7 and 4-8 provide the corresponding information for the regulating rod. It is normally worth less than 200 mbeta. The peak in the differential regulating rod worth occurs at low rod height because the full-in position for the regulating rod is six inches above the bottom of the fuel elements, and because once the regulating rod is withdrawn any appreciable amount, it is heavily shadowed by the adjacent shim blades.

MITR refuelings are normally designed so that the new core will go critical at a shim bank height of 7 to 9 inches. Considerations that limit the shim bank height at which criticality is attained include the subcritical limit interlock and the shutdown margin. At 8.0 inches, the shim bank worth is about 5.85 beta. Equilibrium xenon worth at 6 MW is about 4.2 beta. Hence, the core will operate at a bank height of about 13 inches. The shim bank height will gradually increase as the fuel depletes. Refuelings are normally performed when it is no longer possible to override xenon if the reactor is restarted several hours after a shutdown.

Increases of reactor power are accomplished subject to the following administrative limits:

a) A reactor scram will occur at a period between 10 and 11 seconds.

b) The minimum allowed dynamic period is 30 seconds.

c) If reactor power is less than 80% of demanded, the steady period shall be longer than 50 seconds.

d) If the reactor power is within 80% of demanded, the steady period shall be longer than 100 seconds.

Either a shim blade or the regulating rod may be used to accomplish a power adjustment. The choice is at the discretion of the licensed console operator. (Note: The term "steady period" implies a non-zero reactivity and the absence of control device movement. The term "dynamic period" implies the presence of control device movement with or without a non-zero reactivity.)

4.5.1.6 Interactions of Fuel/Moderator/Reflector/Control Devices

The criticality of any given core configuration is a function of certain interactions between some of the core components. These include:

a) Moderator/Reflector Temperature:

The MITR is intentionally under-moderated so that there will be a negative coefficient of reactivity associated with the temperature of both the moderator (coolant) and the reflector. This temperature coefficient of reactivity encompasses two distinct phenomena. The first is the temperature rise of the light water because of an increase in the thermal power output of the reactor core. Any such temperature rise will insert negative reactivity by causing a hardening in the neutron spectrum. The second phenomenon is the heating of the heavy water reflector. Temperature rises of this type add negative reactivity by allowing neutron leakage to increase. This second process lags the temperature rise of the light water in the core proper. The temperature coefficient of reactivity associated with the entire reactor (H_2O and D_2O) heat-up varies from -6 mbeta/°C to -15 mbeta/°C over the normal band of operating temperatures from 25 to 50 °C. This is depicted pictorially in Figure 4-11.

b) Heavy-Water Reflector Dump

The portion of the MITR's heavy water that is above the bottom plane of the fuel elements is the volume of heavy water that may be dumped. The height of this region is approximately two feet. Because the shim blades also operate in the region between the core and the radial heavy water reflector, the reactivity worth of dumping this radial reflector is dependent on the position of the shim bank. This effect is the result of the shadowing influence that the blade bank exerts on the reflector. The reactivity worths of dumping the radial reflector both when the shim blades are 12 inches withdrawn and when they are in their fully-inserted position are given below:

Shim Bank Position	Reactivity Worth of D_2O Dump
12 Inches Withdrawn	$-16\ \beta$
Fully Inserted	$-10.2\ \beta$

These results show that the reactivity worth of dumping the radial heavy water reflector when the shim bank is fully inserted is about two-thirds that of the corresponding value when the bank is withdrawn by 12 inches. Safety considerations dictate that the radial heavy-water reflector be pumped up with the shim bank in the full-inserted position. This ensures that the reactivity insertion for this process will not occur when the reactor is or could go critical.

c) Effects of Coolant and Reflector Leakage

Light water has both higher moderating power and a higher absorption cross-section than does heavy water. In an under-moderated core with a hardened neutron energy spectrum such as the MITR, the effect of the high moderating power of light water outweighs its absorption effects, and, consequently, replacing this coolant in the core with heavy water will result in a negative reactivity effect. However, in the exterior regions of the light-water core tank, where the neutron energy spectrum is much softer, the absorption by light water outweighs the effect of its high moderating power so that replacing the light water with D_2O in the plenum above the fuel elements and along the walls and bottom of the core tank results in a momentary positive reactivity effect which is followed by a negative effect as the D_2O enters the core proper. Replacement of the heavy water in the reflector tank with light water will always result in a negative reactivity effect.

(i) Leakage of Light Water into the Heavy-Water Reflector

The effect of the leakage of light water into the heavy-water reflector is shown in Figure 4-12. As discussed above, the effect is a negative insertion of reactivity. This change is calculated to be approximately linearly dependent with the percentage of light water contaminant in the heavy-water reflector. It should be noted that the calculation assumes homogeneous mixing of the light-water contaminant with the heavy water in the reflector region. Local effects of H_2O leakage cannot, therefore, be inferred from this data.

(ii) Leakage of Heavy Water into the Light-Water System

The effects of heavy water leakage into the light water system are shown in Figure 4-13. The leakage of pure, uncontaminated heavy water into either the light water above or below the core, or the light-water annular space between the core and the sides of the core tank, results in a positive reactivity effect. On the other hand, leakage of heavy water into the light water of the core proper results in a strong negative reactivity effect. Similarly, progressively replacing the entire light-water system with in-leaking heavy water will produce a strong negative reactivity effect.

Three points should be noted about the leakage of heavy water into the light-water system. First, the calculations were based on the assumption of homogeneous mixing of pure D_2O contaminant with light water in the various regions listed in the figure. Hence, local effects, such as a small pinhole leak, are not directly inferable from this data. Second, the possibility of pure D_2O contaminating any region of the light-water system through a leak in the core tank is highly remote considering that the H_2O side of the system is the higher pressure side. Third, the more likely occurrence is the situation in which the light water from the core tank leaks into the D_2O reflector region until the pressure is equalized on both sides of the core tank. At this point, it is quite possible to have the H_2O-contaminated heavy water diffuse, through the leak in the core tank, into the annular light water regions described above. This situation is illustrated in Figure 4-14 for two homogeneous contaminated D_2O mixtures, one mixture consisting of 10% H_2O - 90% D_2O and the other 30% H_2O - 70% D_2O, diffusing into the annular light-water regions exterior to the core. This calculation shows that an initial negative reactivity effect gradually dissipates until it becomes positive. In the case of the 10% H_2O - 90% D_2O mixture, the turnaround in the reactivity effect occurs when 35% of the light water in the annular light-water region is displaced by the contaminated mixture. As for the 30% H_2O - 70% D_2O mixture, this effect occurs at a 75% displacement. These figures assume that the mixture remains in the annular region. In reality, it would move into the core proper and generate a strong negative reactivity effect as shown in Figure 4-13.

d) In-Core Facilities

There are no current plans for any in-core experimental facilities that would be moved, or in which samples would be moved, during reactor operation. If such a facility or samples were planned, they would be designed to comply with the "movable" reactivity limit as discussed in Section 10.3.2.6 of this report.

4.5.1.7 Safety Considerations for Different Core Configurations

The approach utilized to ensure the safety of every MITR core configuration is discussed in the introduction to Section 4.5 of this report.

4.5.1.8 Reactivity Worths

Reactivity data on MITR fuel, void coefficients, and in-core facilities was estimated by calculation prior to the startup of the MITR-II in 1975. This data was subsequently confirmed by measurement. Much of it is documented in the MITR-II Startup Report [4-12]. Table 4-5 lists some of the more important items.

4.5.1.9 Core Reactivities

Core reactivities are not established in advance because there is no set of pre-defined core configurations. Instead, the following analysis is performed and documented for every planned core in order to be certain that the core is appropriately configured:

a) The expected reactivity change for the proposed refueling is calculated. This can be done by comparing the k-effective values from the CITATION or MCNP calculations of the core before and after the refueling is done in the model. Alternatively, the expected reactivity change may be derived by combining estimates of the change in grams of U-235 with the reactivity coefficients given in Part A of Table 4-5.

b) The shutdown margin for the refueled core is calculated and verified to be acceptable.

c) Items (a) and (b) above are documented and reviewed by someone other than the individual who did the calculations.

d) The refueling is performed.

e) The reactivity change is measured and compared to the predicted estimate.

f) The shutdown margin is recalculated using the measured reactivity change. It is again verified to be acceptable.

g) Items (e) and (f) are documented and reviewed.

Table 4-5

Reactivity Worth Data

A. Fuel

Change	$\Delta\rho$(mβ/g U-235)
	+5.3
	+4.5
	+2.0

B. Void Coefficients

Change	α_v
A-ring: Full channel average	-2.73 mβ/cm^3
B-ring: Full channel average	-2.72 mβ/cm^3
C-ring: Full channel average	-1.64 mβ/cm^3
A-ring: Bottom six inches of channel	-3.47 mβ/cm^3
B-ring: Bottom six inches of channel	-3.53 mβ/cm^3
C-ring: Bottom six inches of channel	-1.88 mβ/cm^3
Core housing (corner hole - downcomer shroud)	-0.28 mβ/cm^3
D_2O blister tank (completely drained)	-70 mβ
Graphite reflector (3GV)	-4.6 mβ/liter

C. Experimental Facilities

	Change	$\Delta\rho$	$\Delta K/K$
Fission Converter		**To be Measured**	
Medical Room	H_2O Shutter	3.4 mβ	2.7×10^{-5}
Below Core	Boral Shutter	0	0
	Lead Shutter	0	0
	D_2O Blister	70 mβ	5.5×10^{-4}

D. Materials in Experimental Facilities

Material	Facility	$\Delta\rho$
	A-ring ICSA	+9.7 mβ/g
	B-ring ICSA	+5.5 mβ/g
	C-ring ICSA	+4.7 mβ/g
	6RH2-2PH1	+7.0 mβ/g
Cadmium	1PH2, 1PH4	0
Cadmium	3GV6	-0.22 mβ/cm^2
Cadmium	6RH2-2PH1	-1.2 mβ/cm^2
Polyethylene	3GV (Bottom)	+0.075 mβ/cm^3
	6RH2-2PH1	+0.012 mβ/cm^3
Graphite	3GV (Bottom)	+0.0046 mβ/cm^3

4.5.1.10 Administrative and Physical Constraints

Both administrative and physical constraints preclude the inadvertent addition of positive reactivity.

a) Administrative: Movement of fuel is not permitted in the core, fuel storage pool, or fission converter without the prior written approval of the Reactor Superintendent or his designate. All such approvals include a schedule of the authorized moves.

b) Physical: The k-effective of all areas where fuel may be stored (except the core itself) is less than 0.9. The principal concern is therefore the core. Changes to the core configuration are not possible when the reactor is operating because of the grid-latch mechanical interlock. In order to obtain a "reactor start" condition, the upper grid plate must be in the latched position. When the grid is latched, no fuel element positions are accessible. That is, fuel can neither be inserted nor removed. This eliminates movement of fuel as a means of inserting positive reactivity during reactor operation. That leaves the possibility of manipulating a sample in an in-core sample assembly (ICSA). All manipulations are scheduled in writing by the Superintendent of Reactor Operations and Maintenance or by the Superintendent's designate. Also, ICSAs are not accessible unless either a reactor top lid penetration or other equivalent shielding is removed. This requires both use of the overhead crane and access to the reactor top. Use of the former is under the control of the licensed console operator. Entry to the latter triggers an alarm in the control room.

4.5.2 Reactor Core Physics Parameters

4.5.2.1 Neutron Lifetime and Effective Delayed Neutron Fraction

Values for both the neutron lifetime and the effective delayed neutron fraction for the MITR-II reactor were estimated prior to the modification in 1974-1975. These values were confirmed by both the dropped rod method and noise analysis as part of the startup testing [4-12]. The prompt lifetime is taken as 100 μs and the effective delayed neutron fraction as 0.00786. The former is typical of a reactor with a thermal neutron spectrum. The latter is higher than that of most other reactors. This is the result of two factors. First, the heavy-water reflector is the source of a significant photoneutron flux that adds to the delayed neutron population that originates from precursor decay. Second, the geometry of the MITR core results in relatively little attenuation of photons within the core. Hence, the photon fraction that interacts with deuterium nuclei in the heavy water is greater than in most other heavy-water reflected cores.

For large reactors, the effective delayed neutron fraction would decrease as the fuel depletes. This effect occurs because a depleted core is effectively larger (blades further withdrawn) in order to sustain criticality. As a result, fewer neutrons are lost to leakage and the relative change in the prompt and delayed neutron populations is lessened during the slowing-down process. For small reactors, this effect is less noticeable. It has not been observed in the MITR. A second factor that may cause the effective delayed neutron fraction to change is the buildup of Pu-239. This also has not been observed for the MITR because the fuel is highly enriched.

4.5.2.2 Coefficients of Reactivity

Values and signs for coefficients of reactivity were given in Table 4-5 of this report. Most of these coefficients were obtained during the startup testing of the MITR-II. Others have been measured as part of special test programs that are required prior to experiment installation. The test method used is as follows:

a) Reactor operation is restricted to low power so that there is no xenon production and/or temperature-dependent reactivity feedback.

b) The reactor is taken critical and critical data (shim blade and regulating rod height, temperature) recorded.

c) The reactor is shut down.

d) The change that is to be investigated is made. For example, a shutter might be opened or a portion of an in-core sample facility voided.

e) The reactor is again taken critical and critical data is recorded.

f) The reactivity worth of the change is determined by comparison of the critical data.

No change in the data shown in Table 4-5 has been noted with fuel burnup with the exception of the figures that pertain to fuel. These do vary (± 20%) with the radial core power distribution.

Analyses given in Chapter 13 of this report show that the void and temperature coefficients of reactivity are sufficiently negative to mitigate reactivity insertion transients.

4.5.2.3 Flux Distributions

The MITR-II's axial and radial neutron flux densities were measured as part of the startup testing in 1975 [4-12]. This was done by use of a special element from which the fuel plates were removable. The method used was to irradiate this special element as part of the core configuration that was to be studied. Individual plates would then be removed and placed in a shielded counter so that a specific portion of the plate, such as a particular axial segment, could be counted.

The axial neutron flux densities were re-measured about a year after these plate scans in order to gain experimental confirmation of the effect of shim bank height on the flux profile. These measurements were performed using copper wires with an appropriate correction for epithermal neutron activation [4-12]. These wire scans confirmed the earlier plate scans.

Figures 4-15 to 4-17 show the axial profiles obtained from the plate scans for A-, B-, and C-Ring positions. Figure 4-18 shows the radial power distribution at mid-height. Additional figures are available in Ref. 4-12.

The axial and radial neutron flux density profiles are independent of power level. They depend on the position of the control devices and hence to the extent that core configuration and burnup change the height of the shim bank, they will also change the flux profiles. The principal findings are listed in Section 4.5.1.3 of this report.

Both the plate-scan and the copper wire data have been used to normalize calculations of the flux profiles that were obtained using various codes. This process is described in the introduction to Section 4.5 of this report. At present (1999), the Monte-Carlo code MCNP [4-13] is being evaluated for use in the fuel management of the MITR.

4.5.3 Operating Limits

Safety is ensured by observance of a substantial shutdown margin requirement as well as by specifying a minimum shim bank height (the subcritical limit interlock) below which the reactor is not to be operated in a critical condition.

4.5.3.1 Reactivity Conditions

The discussion that follows references Figure 4-5 which shows the shim bank integral reactivity worth. The figures cited are for purposes of illustration and are not intended as limiting values.

As discussed in Section 4.5.1.5 of this report, MITR refuelings are normally designed so that the new core will go critical at a shim bank height of 7 to 9 inches. Typical figures are 8.0 inches on the shim bank and 5.0 inches on the regulating rod with a coolant/reflector temperature of 20° C. The reactivity inserted to attain criticality is 5.85 beta (shim bank) and 0.11 beta (regulating rod) for a total of 5.96 beta. The total reactivity worths of the shim bank and regulating rod are 12.63 beta and 0.16 beta respectively. Hence, the excess reactivity in this example is 6.83 beta. The reactivity associated with the attainment of the normal operating temperature (50° C) for the coolant and reflector is -0.26 beta (Figure 4-11), and that associated with the estimated equilibrium xenon is -4.2 beta. A typical in-core sample assembly (secured) might have a reactivity of -0.60 beta, although this figure could be greater (more negative). Also, there could be, and sometimes are, more than one such assembly installed at a time. Hence, assuming only the one ICSA, the excess reactivity remaining once equilibrium operating conditions have been attained is 1.77 beta. Fuel depletion accounts for 0.25 mbeta/MWH. Hence, if no reactivity were needed to override peak xenon, the core could be operated for 7080 MWH or 49 days at 6.0 MW. This is a rather short operating cycle and suggests that the excess reactivity available for MITR operation is quite modest.

4.5.3.2 Excess Reactivity

Excess reactivity and the effect of temperature, xenon, and experiments on excess reactivity is discussed in the preceding section of this report. A typical value for the MITR's excess reactivity is 6 to 7 beta. However, it could be larger because the integral worth of the control devices (shim blades and regulating rod) is 12.8 beta. Two provisions in the MITR's design and operation limit the excess reactivity that is potentially available. First, the subcritical interlock, as described in Section 7.3.1.2 of this report, is observed during startups. Specifically, if the reactor should attain criticality at a shim bank height of less than 5.0 inches, it is to be shut down. The requirement has the effect of reducing the maximum possible excess reactivity from about 12.8 beta to about 9.4 beta. Second, only one shim blade can be withdrawn at a time. This fact, coupled with the identical nature of the six blades and the requirement that they be operated as a bank when above the subcritical interlock, limits the excess reactivity per blade to about 1.6 beta. This is the amount that could, in a worst case scenario, be inserted.

4.5.3.3 Shutdown Margin

The shutdown margin requirement for the MITR is that it be possible to shut the reactor down by at least 1% ΔK/K using shim blades from the cold (10° C), xenon-free condition with both the most reactive blade and the regulating rod fully withdrawn and all movable samples in their most reactive state. For purposes of illustration, assume that the reactor had been operating continuously for a long time, was shut down for 60 hours, and then restarted. It attains criticality with a shim bank height of 8.0" and a regulating rod height of 5.0". The coolant/reflector average temperature is 40° C. There is one in-core sample assembly installed with a worth of -0.60 beta. The shutdown margin is therefore:

a)	Reactivity to attain criticality i) Shim Bank @ 8.0" ii) Regulating rod @ 5.0"	5.96 beta
b)	Worth of most reactive blade	-2.26
c)	Worth of regulating rod	-0.16
d)	Temperature (40 C - 10 C)	-0.13
e)	Xenon (60 hours decay)	-0.28
f)	Samples	-0.60
g)	Shutdown Margin	2.53 beta

The shutdown margin is 2.53 beta. The requirement is 1% ΔK/K which equates to 1.27 beta for the MITR. Thus, the excess shutdown margin is 1.26 beta for this example.

The availability of normal electric power (or for that matter emergency electric power) has no effect on the capability of the MITR to meet the shutdown margin. As discussed in Section 4.2.2.7 of this report, the blades drop into the core under the influence of gravity upon loss of power to electromagnets.

As noted in Section 4.5.1.9 of this report, the shutdown margin is determined prior to every refueling. It is verified following every refueling using the measured worth of the refueling.

The error bar of the shutdown margin calculation is the result of uncertainties associated with the measurement of the individual reactivities that are used in the computation. These in turn reflect the accuracy with which the control device, temperature, and xenon reactivity worth curves were determined as well as the correctness of measurements of the blade and rod height, coolant and reflector temperature, and the time since shutdown. An error analysis of these factors was performed as part of the fuel management studies for the MITR [4-8]. The MITR uses the power doubling time method once asymptotic conditions have been established to measure the reactivity of the control devices. This method is believed accurate to be within ± 10%.

4.5.3.4 Limiting Core Configuration for Thermal-Hydraulic Analysis

Neutron flux and thermal power density increases with the number of in-core sample assemblies. The MITR-II has been operated with two, three, four, and five non-fueled positions. In each case, analysis showed that all thermal-hydraulic and nuclear safety limits were met. The peaking factors used in the thermal-hydraulic analysis contained in this report were obtained on the assumption that there would be five non-fueled positions.

The power peaking factors are a function of both the number of non-fueled positions and the shim bank height. As described in Section 4.5.1.3 of this report, peaking is greater in the inner fuel rings at low bank height. The peaking shifts to the outer fuel ring as the bank is withdrawn. Other factors that affect peaking include the location of the non-fueled positions within the core, the burnup of each element, and the presence of in-core sample assemblies. Accordingly, the core with five non-fueled positions was analyzed at a series of bank heights and the worst-case peaking factors were identified. These factors were then used for all analyses given in Section 4.6 of this report. The factors selected were a radial one of 2.0 and an axial one of 2.1.

Operation of the MITR is normally limited to cases where all six blades are within 2 inches of the bank average. Inoperable blades are a rarity. If an inoperable blade is below the bank average, then the reactor is not to be operated at power levels in excess of 1 kW. This precludes the possibility of a skewed flux profile that might alter the chosen peaking factors. Section 13.2.9.1 reports the analysis of a shim blade drop accident. Operation with an inoperable blade at or above the bank average is permitted because this will not significantly alter the flux density distribution.

4.5.3.5 Transient Analysis

The continuous withdrawal of the most reactive shim blade would create a ramp reactivity addition. Failure of an in-core experiment could create either a ramp or a step reactivity insertion depending on the cause of the failure. Both ramp and step reactivity additions are analyzed in Section 13.2.2 of this report. In neither case is the reactor damaged or fuel integrity lost.

4.5.3.6 Redundancy and Diversity of Reactor Shutdown Mechanisms

The MITR's shutdown mechanisms are both redundant and diverse. Redundancy is provided by six identical shim blades that operate in parallel. Given that the blades are operated as a bank, all blades have essentially the same reactivity worth. Hence, the insertion of any one blade will make the reactor subcritical. Diversity is provided by a completely separate means of shutting the MITR down. This is the capability to dump the portion of the heavy water reflector that is opposite the fuel to a tank located well below the core.

Electricity is required to energize both the electromagnets that support the blades and the solenoid that controls the air pressure that keeps the reflector dump valve closed. Hence, on loss of electricity, the MITR automatically shuts down.

4.6 Thermal-Hydraulic Design

The thermal-hydraulic design of the MITR is described in this section. The MITR may be operated under either forced or natural-convection flow and both modes are addressed here. The MITR does not have a pulse capability and hence that topic is not discussed.

4.6.1 Design Basis

The basis for the MITR's thermal-hydraulic design is that, under conditions of forced convection, the primary coolant system can remove the energy produced during routine 6.0 MW operation of the reactor and transfer it to the secondary coolant system without the onset of nucleate boiling. Another design feature is that the system can also remove at least 100 kW of heat from the fuel elements by natural convection without the onset of nucleate boiling. Provisions are also taken into account in the coolant system design so that fuel integrity is maintained during all credible transients, such as a loss of primary coolant flow because of a pump coast-down (Section 13.2.4). Sufficient margin for the possible deviation of design parameters is taken into account in the thermal-hydraulic limit calculations.

The objective of the thermal-hydraulic design is to guarantee the structural integrity of the fuel elements which are made of a UAl_x matrix enclosed in an aluminum clad. Aluminum melts at approximately 660° C (1200° F). However, it begins to soften significantly at about 450° C (842° F) and the avoidance of this temperature is the criterion adopted here. There are several heat transfer phenomena that could lead to elevated temperatures should they occur. These are:

a) Critical Heat Flux (CHF) at Low Quality: This phenomenon refers to departure from nucleate boiling (DNB). Vapor bubbles form on the fuel clad surface. Initially, this increases heat transfer because of the latent heat that is removed by bubble formation. However, if the heat flux rises, the bubbles coalesce so that patches of vapor exist and heat transfer then decreases because heat must now be conducted through a gas. For a given flow rate, CHF is the heat flux at which this sudden decrease in heat transfer occurs.

b) Critical Heat Flux (CHF) at High Quality: This phenomenon is similar to that described above except that the initiating event is not a departure from nucleate boiling but rather dryout where vapor accumulates in the channel center and gradually strips liquid from the clad surface.

c) Onset of Flow Instabilities (OFI): Flow instabilities refer to the phenomenon where vapor forms in a coolant channel and, as a result of the volume that is suddenly required by the vapor, liquid is displaced. This displacement may result in channel blockage with less coolant flowing to the channel in question and more to adjacent ones. OFI is a possibility in cores with a multi-channel design.

Critical heat flux as the result of departure from nucleate boiling occurs under conditions of high pressure and high heat flux such as exist in pressurized water reactors. Such conditions are not relevant to the MITR. For the MITR, the concerns are CHF as the result of channel dryout and OFI.

Empirical correlations exist to predict both CHF and OFI and both are calculated in Section 4.6.2 of this report in order to establish the MITR's thermal-hydraulic limits.

4.6.1.1 Core Hydraulic Characteristics

The reactor core can be operated with a minimum of 22 fuel elements. The positions that are not occupied by fuel elements can be filled with solid fuel dummies or approved units such as in-core sample assemblies (Section 10.3.2.6) to prevent excessive bypass flow through the non-fueled positions. The MITR can be operated with forced flow or natural convection flow. The former is provided by two primary pumps that are operated in parallel. The nominal flow rate is 2000 gpm. The latter is accomplished by a natural convection flow path in the core tank which is established by the four natural convection valves and the two anti-siphon valves. The natural convection flow is driven by the buoyant force from the coolant exiting the core region. Chapter 6 of this report has a detailed description of the operation of the natural convection and anti-siphon valves.

4.6.1.2 Thermal Power Density Distribution

The thermal power density distribution is assumed to be proportional to the fission density or the neutron flux distribution that are calculated by the MCNP computer code [4-13].

The MCNP code enables full three-dimensional modeling of the exact geometry of the MITR core. The thermal power distribution is a function of fuel depletion, fuel element orientation, and shim bank height. To account for all these parameters in a limit analysis is impractical. Therefore, the power distribution is divided into two independent parameters: radial power peaking factor (F_r) and the axial power peaking factor (F_a). The former is defined as the ratio between the power generated by a fuel plate and the average power per fuel plate. The latter is defined as the ratio between the heat flux for each axial position and the average heat flux along the fuel plate. Thus,

$$F_{r,i} = \frac{P_i}{P_{avg}} \tag{4-2}$$

$$F_{a,z} = \frac{q''(z)}{q''_{avg}} \tag{4-3}$$

where

P_i is power generated by fuel plate i,

P_{avg} is the average power generated in each fuel plate,

$q''(z)$ is the heat flux at axial position z, and

q''_{avg} is the average heat flux along the flux plate.

Figure 4-19 shows F_r as a function of shim bank height. The radial power peaking factors shown in the figure are the maximum values for each shim bank height. Figure 4-20 compares three axial power distributions: bottom-peak, sine/cosine, and uniform. The bottom-peak flux profile was obtained from an MCNP calculation for a shim bank height of eight inches. This low shim bank height results in higher peaking at the bottom of the fuel elements (maximum $F_{a,z}$ of 2.2). The sine/cosine profile is characteristic of a homogeneous core. The uniform flux profile is desirable to achieve uniform fuel burnup and minimum axial peaking.

4.6.2 Major Correlations Used in the Thermal-Hydraulic Limit Calculations

Critical heat flux (CHF) denotes the condition at which the heat transfer of a two-phase flow deteriorates substantially and subsequently leads to elevated fuel temperatures. CHF may occur because of either vapor film formation on the clad surface or dryout of the liquid film. Because the rise in temperature quickly follows CHF, it is desirable to use some other phenomenon, one that occurs earlier, as the basis for a thermal-hydraulic limit. If one slowly increases the energy flux to a heated surface, the first two-phase behavior that is observed is onset of nucleate boiling (ONB), also called incipient boiling. It defines the conditions where bubbles first start to form on the heated surface. Because most of the liquid is still subcooled, the bubbles do not detach but grow and collapse while attached to the wall. ONB is followed by onset of significant voiding (OSV) which describes the condition where the bubbles grow larger on the heated surface and detach regularly. OSV leads to onset of flow instability (OFI) which describes the condition when the mass flow rate decreases with increasing vapor content in the coolant flow. This type of flow excursion will lead to an elevated fuel temperature (or premature CHF) because of an abrupt drop in the coolant flow rate. Experimental studies have shown that OFI tends to occur closely after OSV when the increase in vapor content results in a higher friction pressure drop [4-14].

Correlations are enumerated below for ONB, OFI, and CHF.

4.6.2.1 Correlation for Onset of Nucleate Boiling (ONB)

Sudo *et al.* suggested the Bergles-Rohsenow correlation for the prediction of ONB for narrow rectangular coolant channels [4-15]. This suggestion was based on comparisons of several existing correlations with experimental data. Sudo *et al.* also concluded that the Bergles-Rohsenow correlation predicts the lower limits of the measured ONB temperatures for given heat fluxes and that there exists a margin between the predicted and measured ONB temperatures.

The Bergles-Rohsenow correlation predicts the fuel clad temperature at which ONB occurs.

$$T_{clad,\,ONB} = T_{sat} + 0.556\left[\frac{q''}{1082\,P^{1.156}}\right]^{0.463\,P^{0.0234}} \tag{4-4}$$

where $T_{clad,\,ONB}$ is the fuel clad temperature (°C) at which ONB occurs,

T_{sat} is the saturation temperature (°C),

q'' is the local heat flux (W/m^2 s), and

P is the pressure (bar).

The Bergles-Rohsenow correlation is applicable to both forced convection and natural circulation.

4.6.2.2 Correlation for Onset of Flow Instability (OFI)

Low pressure systems such as the MITR are susceptible to a form of static flow instability known as flow excursion or Ledinegg instability. This type of instability is very important when the gas-to-liquid density ratio is low [4-16]. This flow excursion instability can be described in terms of a plot of channel pressure drop versus mass flow rate as shown in Figure 4-21. The two-phase flow region with a negative slope is the unstable region. Were the hot channel to be operated in this region at point **B**, it is possible that the operating condition would be shifted to point **B'** by a small perturbation. Operation at point **B'** will lead to critical heat flux. Onset of flow instability (OFI) defines the point of inception of this excursion instability.

The point of OFI was determined in previous studies using a steady-state energy conservation equation [4-17]:

$$\dot{m}_{OFI} = \frac{Q}{Rc_p\left(T_{sat} - T_{in}\right)} \tag{4-5}$$

where

$\dot{m}_{OFI}$ is the channel mass flow rate when OFI occurs (kg/s),

Q is the channel power (W),

R is the channel outlet subcooling ratio, $R = (T_{out} - T_{in}) / (T_{sat} - T_{in})$,

c_p is the coolant specific heat (J/kg °C),

T_{sat} is the saturation temperature (°C),

T_{in} is the channel inlet temperature (°C), and

T_{out} is the channel outlet temperature (°C).

The channel outlet subcooling ratio, R, can be determined from the following relations:

$$R = \frac{1}{1 + 25\frac{D_e}{L_h}} \quad \text{, from [4-14]} \tag{4-6}$$

$$R = 0.21 \cdot \ln\left(\frac{L_h}{D_e}\right) - 0.258, \text{ for } 70 < \frac{L_h}{D_e} < 250 \quad \text{, from [4-18]} \tag{4-7}$$

$$R = 0.697 + 0.00063\left(\frac{L_h}{D_e}\right), \text{ for } 100 < \frac{L_h}{D_e} < 200 \quad \text{, from [4-14]} \tag{4-8}$$

where

L_h is the heated length of the coolant channel (m), and

D_e is the equivalent diameter of the coolant channel (m).

The above relations do not include the effect of axial power distribution. This is addressed below by applying the correlation to different power profiles. A uniform one was found to be limiting.

The above correlation for OFI is applicable only to forced convection.

Table 4-6 is a comparison of the calculated mass flow rate at OFI obtained using the above relations and pressure drop calculations. It was assumed in these calculations that the inlet temperature was 50°C and that the flux distribution was uniform. Because Equation (4-8) calculates an OFI flow rate that is more conservative than that obtained from the pressure drop calculation, it is used in the thermal hydraulic calculations of the MITR. Table 4-7 shows the effect of the flux shape on the flow rate at OFI using pressure drop calculations. The calculated results show that a uniform flux distribution will result in a higher OFI flow rate than that obtained

Table 4-6

Comparison of Mass Flow Rates (kg/s) at OFI in the MITR Hot Channel

Power (kW)	Eq.(4-6)	Eq.(4-7)	Eq.(4-8)	Pressure drop calculation
454.0	2.08	2.09	2.21	2.29
227.0	1.04	1.05	1.11	1.11
113.0	0.52	0.52	0.55	0.55
56.8	0.26	0.26	0.28	0.27
40.6*	0.19	0.19	0.20	0.19

* 40.6 kW corresponds to a reactor power of 7 MW with 23 fuel elements and a radial peaking factor of 2.0

Table 4-7

Effect of the Flux Shape on the Flow Rate at OFI (T_{in}=50°C, Q=56.8 kW)

Flux Shape	Flow Rate at OFI (kg/s)
Uniform	0.269
Bottom Peak	0.255

from a bottom-peak flux distribution. The same trend has been observed in previous experiments [4-19]. It is therefore concluded that a uniform flux distribution can be applied in the OFI calculations because it will result in a more conservative estimate of the OFI flow rate or channel heat generation rate. This assumption simplifies the OFI calculation by permitting use of Equations (4-5) and (4-8) instead of pressure drop calculations.

4.6.2.3 Correlation for Critical Heat Flux (CHF)

Critical heat flux (CHF) denotes the conditions at which the heat transfer of the two-phase flow deteriorates substantially and subsequently leads to elevated fuel temperatures. A CHF correlation scheme was proposed for vertical rectangular channels in research reactors by Sudo, *et al.* [4-20]. This correlation scheme is based on experimental data which covers the operating conditions of the MITR. For a forced convection upflow condition, the CHF is calculated as a function of mass flux (G) and the outlet subcooling ($\Delta T_{SUB,o}$). The correlation is:

$$q^*_{CHF,1} = 0.005 \cdot G^{*0.611}\left(1 + \frac{5000}{G^*} \cdot \Delta T^*_{SUB,o}\right) \tag{4-9}$$

Sudo *et al.* also suggested a minimum CHF which corresponds to a zero or blocked flow condition and which can be used to analyze natural convection. It was noted experimentally that when the flow becomes stagnant, a situation could develop where a downward movement of water coexisted with an upward flow of bubbles or steam generated in the channel. This is referred to as a flooding condition because the flow channels are submerged in a pool of coolant. [4-16]

$$q^*_{CHF,2} = 0.7 \cdot \frac{A_{XS}}{A_H} \cdot \frac{\sqrt{W/\lambda}}{\left[1 + \left(\rho_f / \rho_g\right)^{0.25}\right]^2} \tag{4-10}$$

where, for both correlations, we have:

$$q^*_{CHF} = \frac{q''_{CHF}}{h_{fg}\sqrt{\lambda g \rho_g (\rho_f - \rho_g)}} \tag{4-11}$$

$$G^* = \frac{G}{\sqrt{\lambda g \rho_g (\rho_f - \rho_g)}}\,, \tag{4-12}$$

$$\Delta T^*_{SUB,o} = \frac{C_{pf}}{h_{fg}} \cdot (T_{sat} - T_{out})\,,\ \text{and} \tag{4-13}$$

$$\lambda = \left[\frac{\sigma}{g \cdot (\rho_f - \rho_g)}\right]^{0.5} \tag{4-14}$$

where:

A_H	is the heat transfer area,
A_{XS}	is the cross section area,
g	is the acceleration due to gravity,
h_{fg}	is the enthalpy difference between fluid and gas phases,
ρ_f and ρ_g	are the fluid and gas densities, and
σ	is the surface tension.

Equations (4-9) and (4-10) can be plotted as shown in Figure 4-22. G^*_0 is the intersection of $q^*_{CHF,1}$ and $q^*_{CHF,2}$. When the mass flux drops below G^*_0, the fuel will be cooled by countercurrent flow (flooding). For the geometry of the MITR fuel elements, G_0 is calculated to be 1.34 kg/m^2 s, which is less than 0.1% of normal flow. The calculated $q_{CHF,2}$ is 2.353 x 10^4 W/m^2 s, which corresponds to a reactor power of 468 kW with a radial peaking factor of 2.0.

The CHF correlation is applicable to forced convection ($q^*_{CHF,1}$) and to natural convection ($q^*_{CHF,2}$).

4.6.3 Reactor Power Deposition and Core Flow Distribution

The correlations described above are to be used for the "hot channel". The hot channel is the coolant channel that presents the most limiting operating conditions, e.g., highest power, least coolant flow, and most severe axial peaking. These conditions may not all occur in the same coolant channel. In fact, it was found from the experience obtained with the MITR-II that these conditions occurred in different channels [4-12]. It is assumed in this study that the hot channel has the combination of all the limiting conditions. This conservative assumption is used to determine the thermal-hydraulic limits that establish the lower bound of acceptable operating conditions. This assumption simplifies the thermal-hydraulic limit calculations to analysis of the hot channel and the average channel.

The flow rate and heat generation in the hot channel are represented as a function of the reactor primary coolant flow rate and reactor power because the latter are the parameters that are measured by the reactor instrumentation. This representation is achieved by using a set of factors that are derived from either calculations or experimental data.

4.6.3.1 Reactor Power Deposition

The energy produced by the reactor consists of the kinetic energy of the fission fragments, beta particle energy, prompt and delayed gamma energy, the kinetic energy of the prompt and delayed neutrons, and that of the neutrinos. Kinetic energy of the fission fragments and most of the beta particle energy is deposited in the fuel because of the short range of these fragments and particles. This portion of the fission energy is about 90% of the total fission energy. A small percentage of the fission power is deposited outside the reactor's core region because of the long mean-free-path of gammas and fast neutrons compared to the small size of the core. Also, neutrino energy is not deposited. The core deposition factor (F_{core}) defines the fraction of the fission power deposited in the reactor core region (both fuel and coolant) of the core tank. The fuel deposition factor (F_{fuel}) defines the fraction of the core power deposited in the fuel elements. The radial power peaking factor (F_r) is described in Section 4.6.1.2 of this report. The power generated in the hot channel can be calculated using

$$Q = \frac{P}{N_c} F_{core} F_r \qquad (4\text{-}15)$$

where P is the reactor power and N_c is the total number of coolant channels in the core. The axial heat flux distribution of the fuel plate adjacent to the hot channel can be calculated as:

$$q''(z) = \frac{Q}{A_H} F_{fuel} \theta(z) \qquad (4\text{-}16)$$

where A_H is the total heat transfer area for one fuel plate and $\theta(z)$ is the normalized axial flux distribution factor.

Measurements performed for the MITR-II showed that about 2.0% of the fission energy is deposited in the reactor's heavy-water reflector system, and about 1.5% in the shield system. MCNP calculations showed that 94% of the energy deposited in the core region is deposited in the fuel elements [4-21]. F_{core} and F_{fuel} are therefore estimated to be 0.965 and 0.940, respectively.

4.6.3.2 Core Flow Distribution

Ideally, the core region should be designed so that 100% of the coolant flows through the fuel elements. However, in reality part of this flow bypasses the core region through the natural circulation and anti-siphon valves, and because of design/manufacturing tolerances, such as clearances, between the fuel elements. The coolant flow factor (Ff) is defined as the ratio of the primary coolant flow that actually cools the core region to the total flow. The MITR-II has a core coolant flow factor of 0.921 which was determined experimentally during the initial startup testing [4-12].

The channel flow disparity factor (d_f) is defined as the ratio of the minimum flow to the average flow in the coolant channels. It is:

$$d_f = \frac{\dot{m}_{min}}{\dot{m}_{avg}} \tag{4-17}$$

where $\dot{m}_{min}$ is the minimum flow rate measured in all the coolant channels and $\dot{m}_{avg}$ is the average flow rate in the coolant channels.

The flow distribution in the reactor core was measured during the MITR-II's initial startup testing. The minimum flow through a fuel element is 93% of the average core flow rate. The flow distribution within a fuel element has also been measured experimentally using a dummy element. The ratio of the minimum channel flow rate to the average channel flow rate within a fuel element is 0.929 [4-12]. Hence, the worst-case channel flow disparity factor of the MITR can be calculated as:

$$d_f = 0.93 \times 0.929 = 0.864 \tag{4-18}$$

The minimum coolant channel flow in the core is then calculated in the thermal hydraulic analysis using the following equation:

$$\dot{m}_{min} = \frac{W_p}{N_c} \cdot F_f \cdot d_f \tag{4-19}$$

where W_p is the total primary coolant flow rate and N_c is the number of coolant channels in the core region.

4.6.4 Engineering Hot Channel Factors

The engineering hot channel factors account for possible deviations from nominal design specifications that may affect the thermal-hydraulic calculations. Specifically, they are defined for channel enthalpy rise, film temperature difference, and heat flux [4-16]. These parameters encompass sub-factors that can be combined either multiplicatively or statistically to obtain the engineering hot channel factors. It has been concluded that it is overly conservative to combine the sub-factors multiplicatively [4-16, 4-22]. So, the statistical approach has been used in the current study. Table 4-8 is a summary of the engineering hot channel factors.

The engineering hot channel sub-factors considered in the current study include those for reactor power measurement, power density measurement/calculation, fuel density tolerances, flow channel tolerances, fuel meat eccentricity, and heat transfer coefficient prediction. Numerical values for these sub-factors were mostly adopted from the MITR-II SAR [4-23].

A "vertical" approach is used in the current analysis to calculate the maximum fuel clad temperature and the maximum coolant temperature using the engineering hot channel factors. This approach is the standard conventional method noted in Ref. [4-16]. To calculate the maximum coolant temperature, use

$$T_{c,M} = T_{in} + F_H \Delta T_c \quad (\text{or } H_{c,M} = H_{in} + F_H \Delta H_c \text{ if boiling occurs}) \tag{4-20}$$

To calculate the maximum fuel clad temperature, use:

$$T_{w,M} = T_{in} + F_H \Delta T_c + F_{\Delta T} \Delta T_w \tag{4-21}$$

where T_{in} is the coolant channel inlet temperature,

H_{in} is the coolant enthalpy at the channel inlet,

ΔT_c is the coolant temperature rise,

ΔH_c is the coolant enthalpy rise,

ΔT_w is the film temperature rise (temperature difference between coolant and clad),

F_H is the engineering hot channel factor for enthalpy rise,

$F_{\Delta T}$ is the engineering hot channel factor for film temperature rise,

$T_{c,M}$ is the maximum coolant temperature because of design / manufacture deviations,

Table 4-8

Engineering Hot Channel Factors Applied in the Thermal-Hydraulic Calculations

Enthalpy Rise			
Reactor power measurement			1.05
Power density measurement/calculation			1.10
Plenum chamber flow			1.08
Flow measurement			1.05
Fuel density tolerances			1.026
Flow channel tolerances			1.089
Eccentricity			1.001
	F_H,	Statistical	1.173
Film Temperature Rise			
Reactor power measurement			1.05
Power density measurement/calculation			1.10
Plenum chamber flow			1.06
Flow measurement			1.04
Fuel density tolerances			1.05
Flow channel tolerances			1.124
Eccentricity			1.003
Heat transfer coefficient			1.200
	$F_{\Delta T}$,	Statistical	1.275
Heat Flux			
Reactor power measurement			1.05
Power density measurement/calculation			1.10
Fuel density tolerances			1.05
Eccentricity			1.003
	F_Q	Statistical	1.123

Note: The engineering hot channel factors are obtained by combining the sub-factors statistically using the equation $F = 1 + \left[\sum_j (f_j - 1)^2\right]^{1/2}$, where f_j denotes sub-factors [Ref. 4-16].

$H_{c,M}$ is the maximum coolant enthalpy because of design / manufacture deviations, and

$T_{w,M}$ is the maximum fuel clad temperature because of design / manufacture deviations.

The quantities ΔT_c (or ΔH_c), and ΔT_w are calculated for every axial node in the hot channel with the operating limits taken into account. $T_{c,M}$ is then calculated and checked if the ONB limits are exceeded.

For OFI, the following equation is used to account for uncertainties:

$$\dot{m}_{OFI} = \frac{F_H \cdot Q}{Rc_p\left(T_{sat} - T_{in}\right)}, \tag{4-22}$$

or

$$Q_{OFI} = \frac{\dot{m}Rc_p\left(T_{sat} - T_{in}\right)}{F_H} \tag{4-23}$$

where $\dot{m}$ and Q are the flow rate and power, respectively, in the hot channel.

For critical heat flux, the following two equations are used:

$$q''_{hs,M} = q''_{avg}F_aF_r\ F_Q \tag{4-24}$$

$$\frac{q''_{CHF}}{q''_{hs,M}} > \text{MCHFR} \tag{4-25}$$

where

q''_{avg} is the core average heat flux,

$q''_{hs,M}$ is the maximum hot spot heat flux because of design manufacture deviations,

q''_{CHF} is the calculated critical heat flux,

$(F_aF_r)_{max}$ is the maximum power peaking factor (radial multiplied by axial),

F_Q	is the engineering hot channel factor for heat flux, and
MCHFR	is the minimum critical heat flux ratio.

The value of MCHFR is determined for the CHF correlation to account for the lower bound of the experimental data used to developed the CHF correlation. It is suggested by Sudo, *et al.*, that MCHFR of 1.5 or higher should be used for Equation (4-25).

4.6.5 Thermal-Hydraulic Limits

The MITR is designed to operate in a safe manner under all credible conditions. The thermal-hydraulic design limits were chosen to provide a safe margin beyond the desired operating range. Calculations of the design limits also include considerations for deviations from design specifications.

The safety limits are established to maintain the integrity of the fuel clad. Critical heat flux (CHF) is normally used as the criterion of fuel overheating. However, because the coolant flow path in the MITR core is a multichannel design, there exists the possibility that flow instabilities could occur before reaching CHF limitations. If onset of flow instability (OFI) did occur first, it would have the effect of lowering the flow rate to the hot channel significantly and thus lowering the critical heat flux. In the safety limit calculations, both CHF (Sudo correlation) and OFI are calculated and the one that would occur first is used to determine the safety limits.

The limiting safety system settings (LSSS) are established to allow a sufficient margin between normal operating conditions and the safety limits. Onset of nucleate boiling (ONB) is chosen as the criterion for the LSSS derivation. This guarantees that boiling will not occur anywhere in the fueled region as long as the limits are not exceeded. Specifically, the LSSS for forced flow operations is set for:

a) The maximum reactor power,

b) The maximum steady-state average core outlet temperature,

c) The minimum primary flow rate, and

d) The minimum coolant level in the core tank.

The LSSS for natural-convection operation is set for:

a) The maximum reactor power,

b) The maximum steady-state average core outlet temperature, and

c) The minimum coolant level in the core tank.

The MITR can operate with all four/three parameters simultaneously approaching their limits without there being boiling in the fueled region.

Calculations for possible transients such as the continuous withdrawal of a control device are given Chapter 13 of this report. Those results show that sufficient time exists between the attainment of an LSSS and the attainment of a safety limit that the MITR's safety system would be capable of negating the transient without damage to the fuel.

4.6.6 Derivation of the Safety Limits

4.6.6.1 Comparison of OFI and CHF

Figure 4-23 shows the calculated critical heat flux ratio (CHFR) for three axial power profiles. The first is a bottom-peak flux profile obtained from an MCNP calculation for a shim bank height of eight inches. This low shim bank height results in a high peaking at the bottom of the fuel elements. The other profiles are sine/cosine (homogeneous core) and uniform. The factors required for the calculation were taken as follows: It is assumed that the reactor power is 7 MW, reactor outlet temperature is 60°C, the hot channel subcooling is 10°C, and the radial peaking factor is 2.0. The figure shows that the minimum heat flux ratio (2.79) occurs at the bottom of the fuel. The bottom peak flux distribution is the limiting case because of its high peaking at the bottom of the fuel.

Figure 4-24 is a comparison of the minimum CHF, obtained from the bottom-peak flux distribution described above, and the OFI heat flux for forced flow conditions. The ratio increases with decreasing mass flux. This ratio is 3.8 at a mass flux 2000 kg/m^{2}s, which corresponds to the mass flux in the coolant channel which receives the least flow at a total primary flow of 1800 gpm. Upon dividing by 2.2 (maximum axial peaking factor), this ratio becomes 1.72. This adjustment is made because CHF is used for local heat flux and OFI is a channel average quantity. This represents an estimate of CHF which is about 70% higher than that of OFI. It is concluded that OFI is a more conservative estimate than CHF for the MITR and therefore OFI should be used as the basis of the safety limits for the MITR instead of CHF. (Note: This conclusion applies only to forced convection because the OFI correlation does not extend to natural convection.)

4.6.6.2 Calculation of the Safety Limits for Forced Convection

It is assumed conservatively that the hot channel (one with the maximum radial peaking factor, F_r) receives the minimum amount of flow among all the coolant channels. This assumption establishes the lower bound of the thermal-hydraulic limits.

The hot channel flow rate and power can be written as functions of total primary flow rate (W_p) and reactor power P using the following equations:

$$\dot{m} = \frac{W_p}{N_c} F_f d_f \tag{4-26}$$

$$Q = \frac{P}{N_c} F_{core} F_r \tag{4-27}$$

Core inlet temperature can be written as a function of core outlet temperature, reactor power, and total primary flow rate.

$$T_{in} = T_{out} - \frac{P \cdot F_{core}}{W_p c_{pf}} \tag{4-28}$$

Upon combining Equations (4-26) through (4-28) with Equation (4-22), the safety limit equation is derived as follows:

$$\frac{P}{W_p} = \frac{c_{pf}\left(T_{sat} - T_{out}\right)}{F_{core}\left(\frac{F_r F_H}{RF_f d_f} - 1\right)} \tag{4-29}$$

The safety limits are calculated assuming R=0.86 (calculated from Equation (4-5)), F_r=2.0, $F_f d_f$= 0.8 , F_H=1.173, and F_{core}=1.0. Hence $F_{core}\left(\frac{F_r F_H}{RF_f d_f} - 1\right)$, denoted as the safety limit factor, equals 2.4. Figure 4-25 shows the safety limits for coolant heights of 10 and 6 feet. The coolant height is the elevation from the top of the fuel plates to the air/water interface on the top of the core tank. A coolant height of 10 feet corresponds to 4 inches below the overflow level. A coolant height of 6 feet corresponds to 5 inches under the anti-siphon valves.

4.6.6.3 Calculation of the Safety Limits for Natural Convection

The safety limit for natural convection operation is calculated using the CHF correlation derived for a zero or blocked flow condition, Equation (4-30). Experiments found that the coolant channels can be cooled by countercurrent flow with a downward water flow and an upward flow of bubbles or steam generated in the channel. This is referred to as a flooding condition because the flow channels are submerged in a pool of coolant. The maximum heat flux corresponding to this condition can be calculated using the following equations [4-15]:

$$q^{*}_{CHF,2} = 0.7 \cdot \frac{A_{XS}}{A_H} \cdot \frac{\sqrt{W/\lambda}}{\left[1 + \left(\rho_f / \rho_g\right)^{0.25}\right]^2} \tag{4-30}$$

and,

$$q^{*}_{CHF} = \frac{q''_{CHF}}{h_{fg}\sqrt{\lambda g \rho_g \left(\rho_f - \rho_g\right)}} \tag{4-31}$$

$$\lambda = \left[\frac{\sigma}{g \cdot (\rho_f - \rho_g)} \right]^{0.5} \tag{4-32}$$

The calculated CHF is 2.353 x 10^4 W/m^2 s, which corresponds to a reactor power of 468 kW with a hot channel factor of 2.0. Upon taking into account the engineering hot channel factor for enthalpy rise (F_H), the reactor power corresponding to a dry-out condition becomes 399 kW. Table 4-9 lists the proposed safety limit for natural convection operation. A reactor power of 350 kW is conservatively adopted as the safety limit. The core outlet temperature and the coolant height do not affect the dryout limit, as long as the core is covered with coolant. The coolant height is conservatively set at 6 ft above the top of the fuel plates to ensure an adequate coolant inventory.

4.6.7 Calculation of the Limiting Safety System Settings

Onset of nucleate boiling (ONB) is chosen as the criterion for the limiting safety system settings (LSSS). This guarantees that boiling will not occur anywhere in the core and that the safety limits will not be approached. The ONB limit is calculated based on the fuel clad temperature. The following equation needs to be satisfied in order to prevent ONB.

$$T_{clad,M}(z) < T_{clad,ONB}(z) \tag{4-33}$$

where $T_{clad,M}$ is the maximum clad temperature defined in Equation (4-20) and $T_{clad,ONB}$ is the clad surface temperature at which ONB occurs. The appropriate thermal properties are those for 1.3 bar, which corresponds to a coolant height of 10 feet (3 meters) above the top of the fuel plates or 4" below overflow. Equation (4-4) becomes:

$$T_{clad,ONB} = 107 + 0.0177 q''(z)^{0.466} \tag{4-34}$$

Table 4-9

Proposed Safety Limits for Natural Convection Operation.

Variable	Safety Limits (0 pumps)
Power	350 kW (max)
Coolant Height	6 feet above top of fuel plates

$T_{clad,M}$ is the maximum clad surface temperature obtained using the "vertical" approach as described in Section 4.6.4 of this report.

$$T_{clad,M}(z) = T_{in} + \frac{F_H}{\dot{m}c_{pf}} \int_0^z P_H q''(z)dz + F_{\Delta T}\frac{q''(z)}{h} \quad (4\text{-}35)$$

P_H is the heat transfer perimeter of a coolant channel. T_{in} can be written as a function of reactor power and core outlet temperature using the following equation:

$$T_{in} = T_{out} - \frac{P \cdot F_{core}}{W_p c_{pf}} \quad (4\text{-}36)$$

Upon combining Equation (4-33) through (4-35), Equation (4-36) becomes,

$$T_{out} - \frac{P \cdot F_{core}}{W_p c_{pf}} + \frac{F_H}{\dot{m}c_{pf}} \int_0^z P_H q''(z)dz + F_{\Delta T}\frac{q''(z)}{h} - 107 - 0.0177q''(z)^{0.466} < 0 \quad (4\text{-}37)$$

The left side of the equation is denoted as ΔT_{ONB}, which is the subcooling of the coolant.

The heat flux can be calculated using

$$q''(z) = \frac{Q}{A_H} F_{fuel}\theta(z), \text{ and} \quad (4\text{-}38)$$

$$Q = \frac{P}{N_c} F_{core} F_r \quad (4\text{-}39)$$

Figure 4-26 is a comparison of ΔT_{ONB} for the same three axial power distributions that were used in Section 4.6.6.1 of this report. A reactor power of 7 MW, a reactor outlet temperature at 60°C, a hot channel subcooling of 10°C, and a radial peaking factor of 2.0 are again assumed. The results show that the uniform power distribution results in the minimum ΔT_{ONB}, which occurs at the outlet of the coolant channel. Although the peaking at the inlet of the fuel for the bottom-peak flux distribution has a notable effect on ΔT_{ONB}, the minimum ΔT_{ONB} for the bottom-peak power distribution occurs close to the mid-height of the coolant channel. It is therefore concluded that the uniform power distribution is the limiting case for the ONB calculation. Equation (4-37) can be simplified for a uniform power distribution as follows:

$$T_{out} - \frac{P \cdot F_{core}}{W_p c_{pf}} + \frac{F_H}{\dot{m} c_{pf}} q''_{avg} A_H + F_{\Delta T} \frac{q''_{avg}}{h} - 107 - 0.0177 {q''_{avg}}^{0.466} < 0 \qquad (4\text{-}40)$$

where

$$q''_{avg} = \frac{P}{N_c A_H} F_{fuel} F_{core} F_r \qquad (4\text{-}41)$$

4.6.7.1 Calculation of the Limiting Safety System Settings for Forced Convection

Figures 4-27 and 4-28 are the calculated LSSS for two-loop operation and one-loop operation, respectively, that are obtained by using Equation (4-40). Two loop operation uses a LSSS primary flow rate of 1800 gpm, while one loop uses 900 gpm. The coolant height is set at 10 feet above the top of the fuel plates, or 4 inches below overflow, for both cases. Table 4-10 lists the proposed instrument settings for two-loop and one-loop operation.

4.6.7.2 Calculation of the Limiting Safety System Settings for Natural Convection

Natural convection calculations are made using the MULCH-II code [4-25]. The calculations were performed assuming that the reactor is at 1 MW before a loss of primary flow occurs and that the reactor power remains at 100 kW. This approach ensures that the steady-state results are conservative. Natural convection flow establishes itself after the forced flow stops. The coolant height is assumed to be at 10 feet above the top of the fuel plates, or 4 inches below

Table 4-10

Limiting Safety System Settings for Forced-Convection Operation

Parameter	LSSS (2 loops)	LSSS (1 loop)
Power	7.4 MW (max)	3.2 MW (max)
Primary Coolant Flow	1800 gpm (min)	900 gpm (min)
Steady-State Average Core Outlet Temperature	60 °C (max)	60 °C (max)
Coolant Height	4" below overflow (min), or 10 feet above top of fuel plates	4" below overflow (min), or 10 feet above top of fuel plates

overflow. The radial power peaking factor (or nuclear hot channel factor) is assumed to be 2.0. Calculations were done for two cases: both natural convection valves (NCV) and anti-siphon valves (ASV) operable, and only NCV operable. Descriptions of the design and operation of the NCVs and ASVs are given in Sections 6.2 and 6.3 of this report. The first case is applicable to low-power operation without forced primary flow. The second case may occur if the coolant level drops below the ASV (about 6.4 ft above top of the core). The natural convection flow rate approaches a constant at about 80 seconds. The maximum fuel clad temperature can be calculated using equation (4-21), which was given in Section 4.6.4. F_H and $F_{\Delta T}$ are 1.173 and 1.275 respectively from Table 4-8. Table 4-11 lists the natural convection flow rate through the fueled region, the coolant and the clad temperature rise for the hot channel, the maximum coolant and clad temperature rises using the hot channel factors, and finally the maximum clad temperature assuming an inlet temperature at 60°C. It is assumed conservatively that the core inlet temperature is equal to the LSSS core outlet setpoint. This assumption simplifies the calculation by neglecting the transient time for the natural convection coolant flow to go from the upper plenum to the core inlet. The maximum clad temperatures are 104 °C and 105 °C for case 1 and 2, respectively. The saturation temperature for a coolant height of 4 inches below overflow is about 107 °C. Therefore, there will be no boiling in the core. Table 4-12 is the proposed limiting safety system settings for natural convection operation.

4.6.8 Refueling Considerations

There is no set of pre-planned core configurations for the MITR. Rather, any configuration is acceptable provided that certain criteria are met. These include:

a) Each of the twenty-seven positions within the core must contain a fuel element, or a solid aluminum dummy, or an approved in-core sample assembly (ICSA). There should be at least 22 fuel elements in the core. (Note: Also acceptable is a neutron source tube as discussed in Section 4.2.4 of this report.)

b) Refuelings can be performed by replacement of spent fuel with fresh or partially used fuel, shuffling, rotation, and inversion (flipping) of individual elements to negate the effect of radial and axial flux gradients.

Table 4-11

Calculated Coolant Temperature Rise and
Film Temperature Rise for Natural Convection Operation

Parameter	NCV&ASV	NCV only
Flow rate through fueled region (kg/s)	0.97	0.93
ΔT_c (°C)	33.56	34.02
ΔT_w (°C)	3.62	3.61
T_w (°C) (T_{in} =60°C)	97.18	97.63
$\Delta T_{c,M}$ (°C)	39.37	39.91
$\Delta T_{w,M}$ (°C)	4.62	4.60
$T_{w,M}$ (°C) (T_{in} =60°C)	103.99	104.51

Table 4-12

Limiting Safety System Settings for Natural Convection Operation

Parameter	**LSSS (0 pumps)**
Power	100 kW (max)
Steady-State Average Core Outlet Temperature	60 °C (max)
Coolant Height	4" below overflow (min), or 10 feet above top of fuel plates

c) Certain thermal-hydraulic parameters must not be exceeded for any channel in the core. These include the safety limit factor, $F_{core}\left(\frac{F_r F_H}{RF_f d_f} - 1\right)$, and onset of nucleate boiling, Eq. (4-37).

d) No fuel element, or any portion thereof, can exceed the fission density limit during the projected operation cycle.

e) The shutdown margin requirement is met. The expected reactivity change for the proposed refueling is calculated. This can be done by comparing the k-effective from calculations of the core before and after the refueling is done in the model. Or, it could be done by combining estimates of the change in grams of U-235 with the reactivity coefficients given in Part A of Table 4-5.

f) Items (c) through (e) above are documented and reviewed by someone other than the individual who did the calculations. Each movement of the fuel elements is planned so the only one fuel element is moved at a time.

g) The refueling is performed.

h) The reactivity change is measured and compared to the predicted estimate.

i) The shutdown margin is recalculated using the measured reactivity change. It is again verified to be acceptable.

j) Items (h) and (i) are documented and reviewed.

If any fuel element is to be moved to the spent fuel pool, requirements are imposed to limit decay heat before the fuel transfer to prevent excessive temperature.

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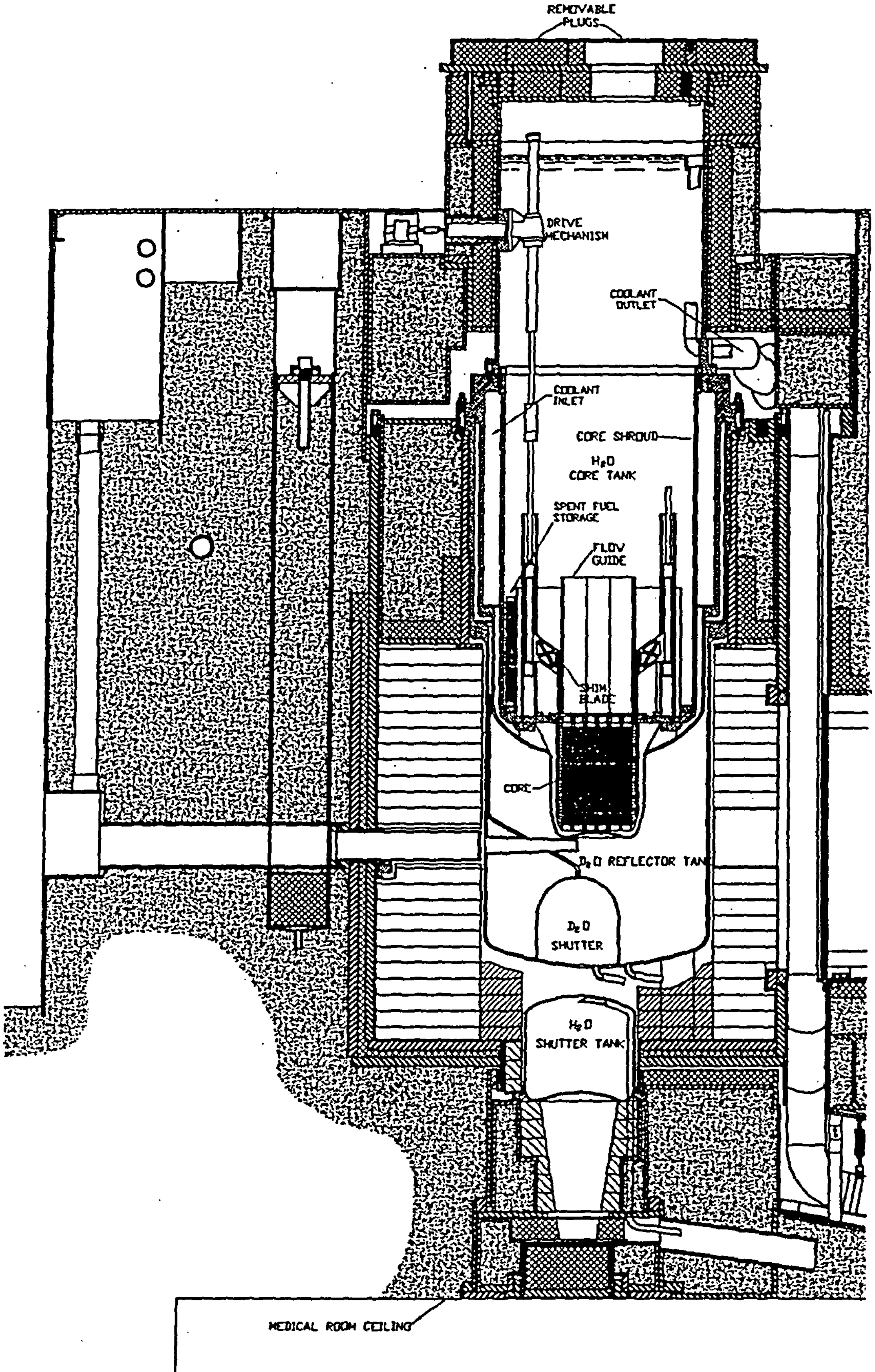


FIGURE 4-1
VERTICAL CROSS-SECTION MITR

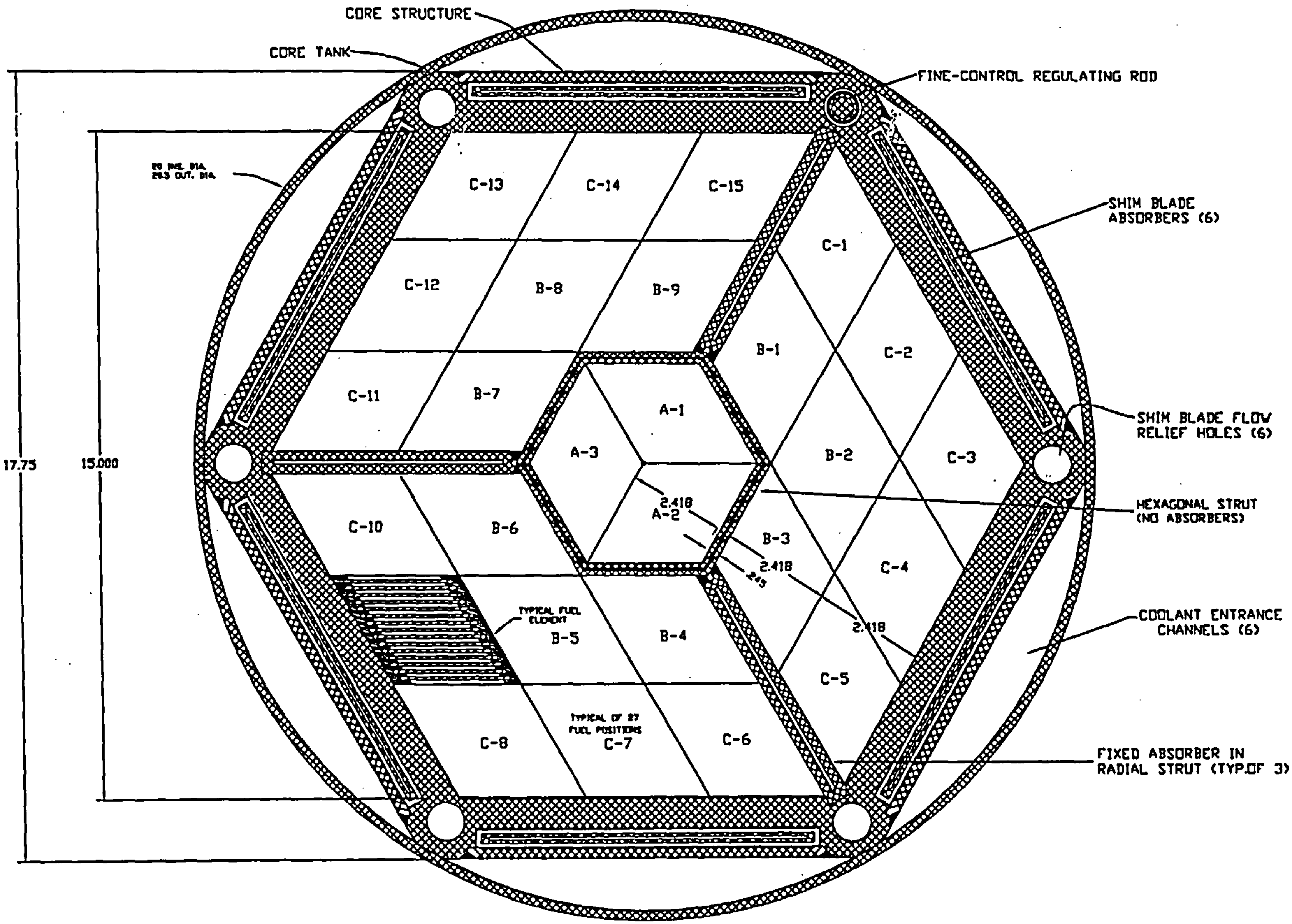


FIGURE 4-2 CORE SECTION MITR

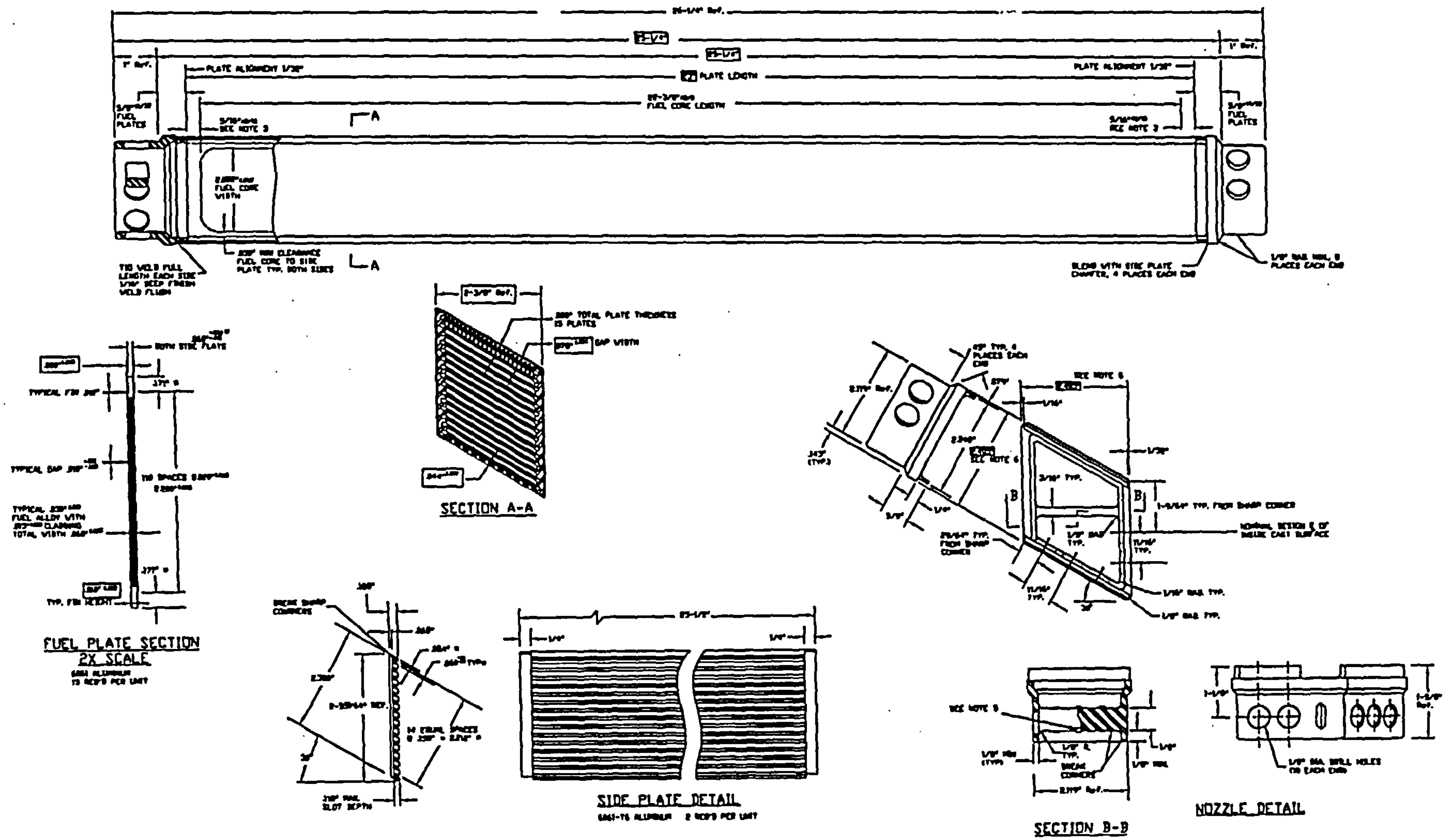
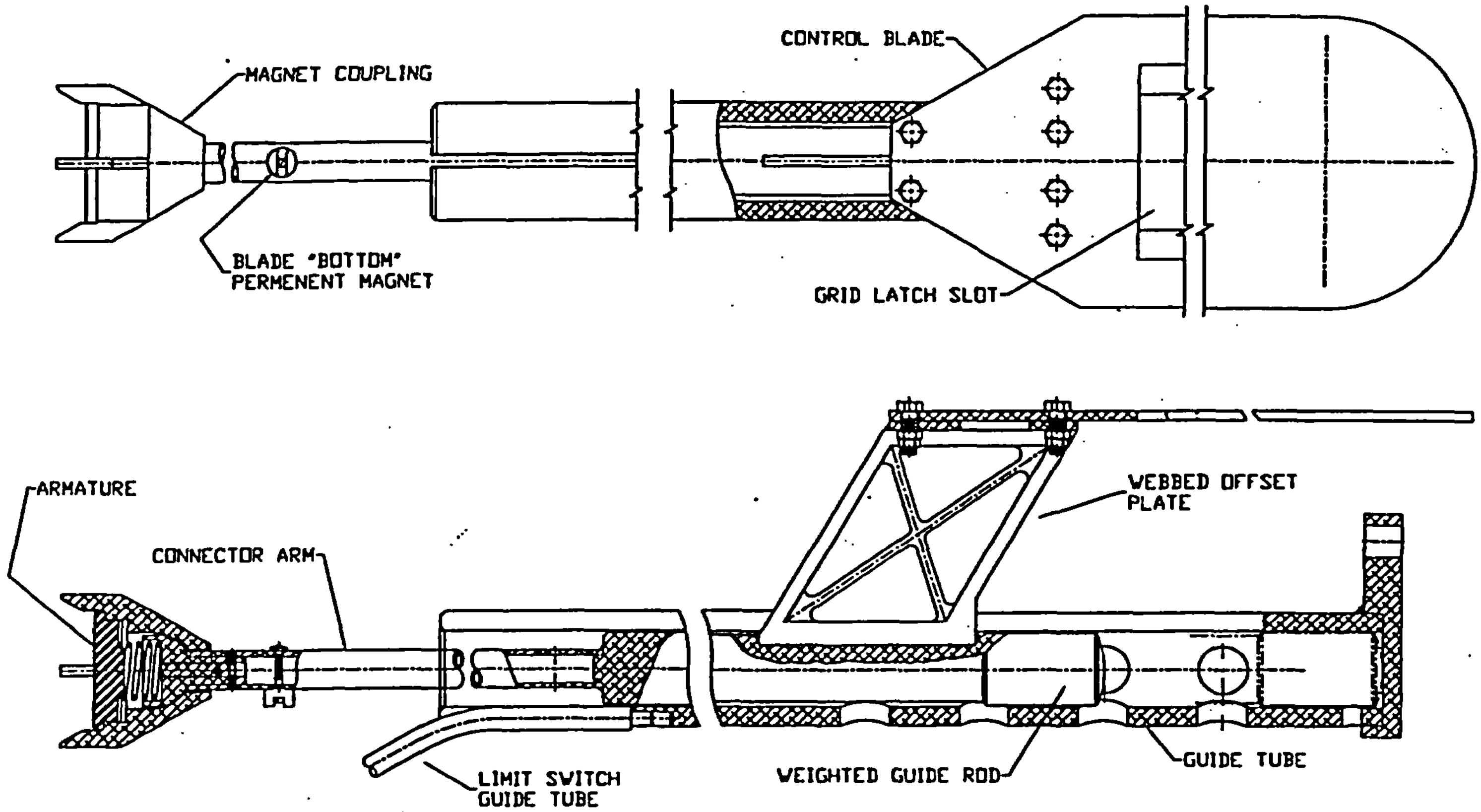
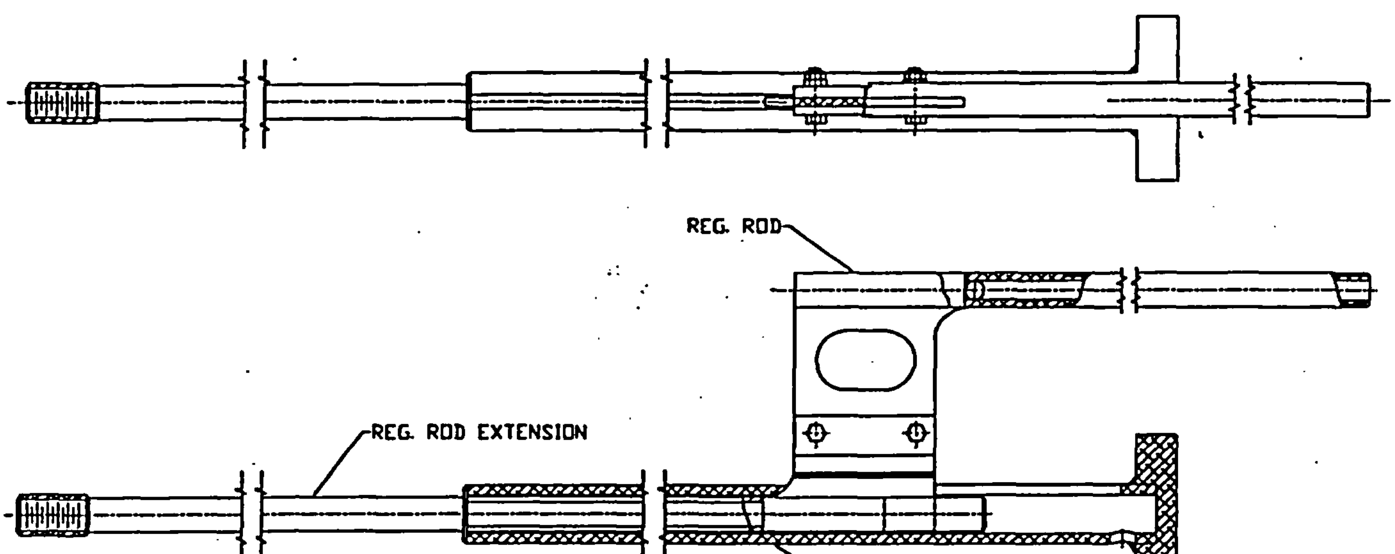


FIGURE 4-3
MITR3 FUEL ELEMENT



CONTROL BLADE ASSEMBLY



REGULATING ROD ASSEMBLY

FIGURE 4-4

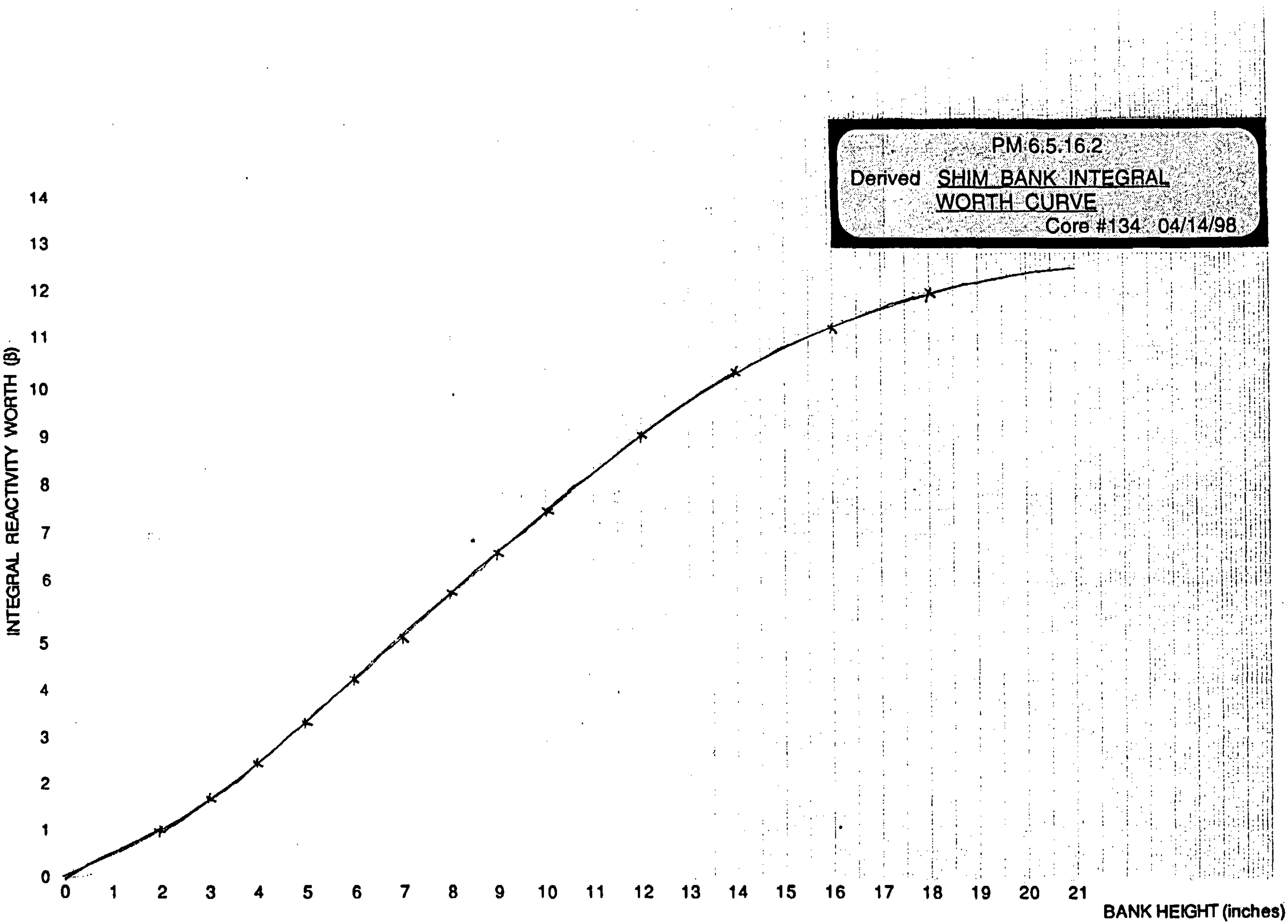


Figure 4-5 MITR Shim Bank Integral Curve.

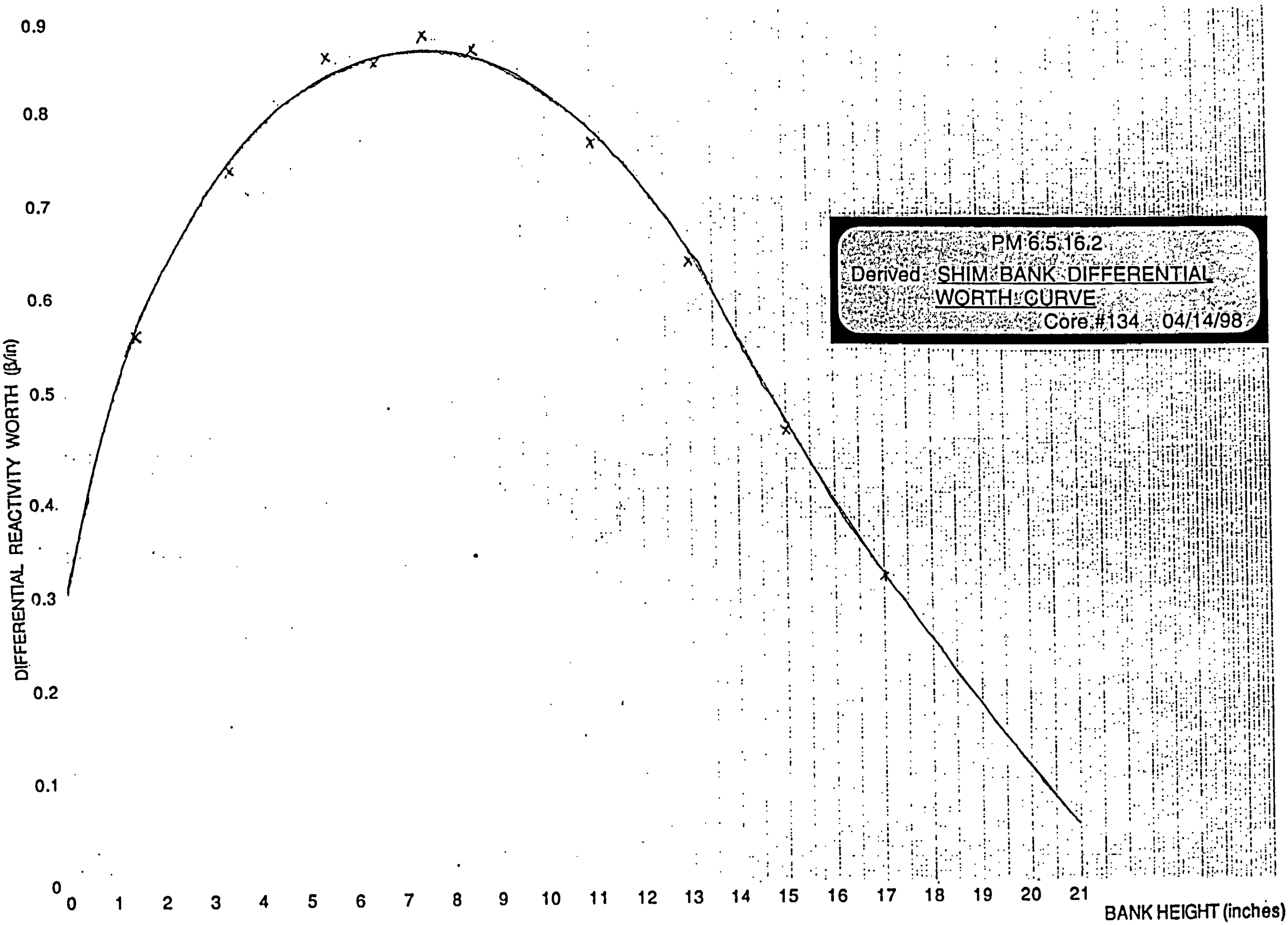


Figure 4-6 MITR Shim Bank Differential Curve.

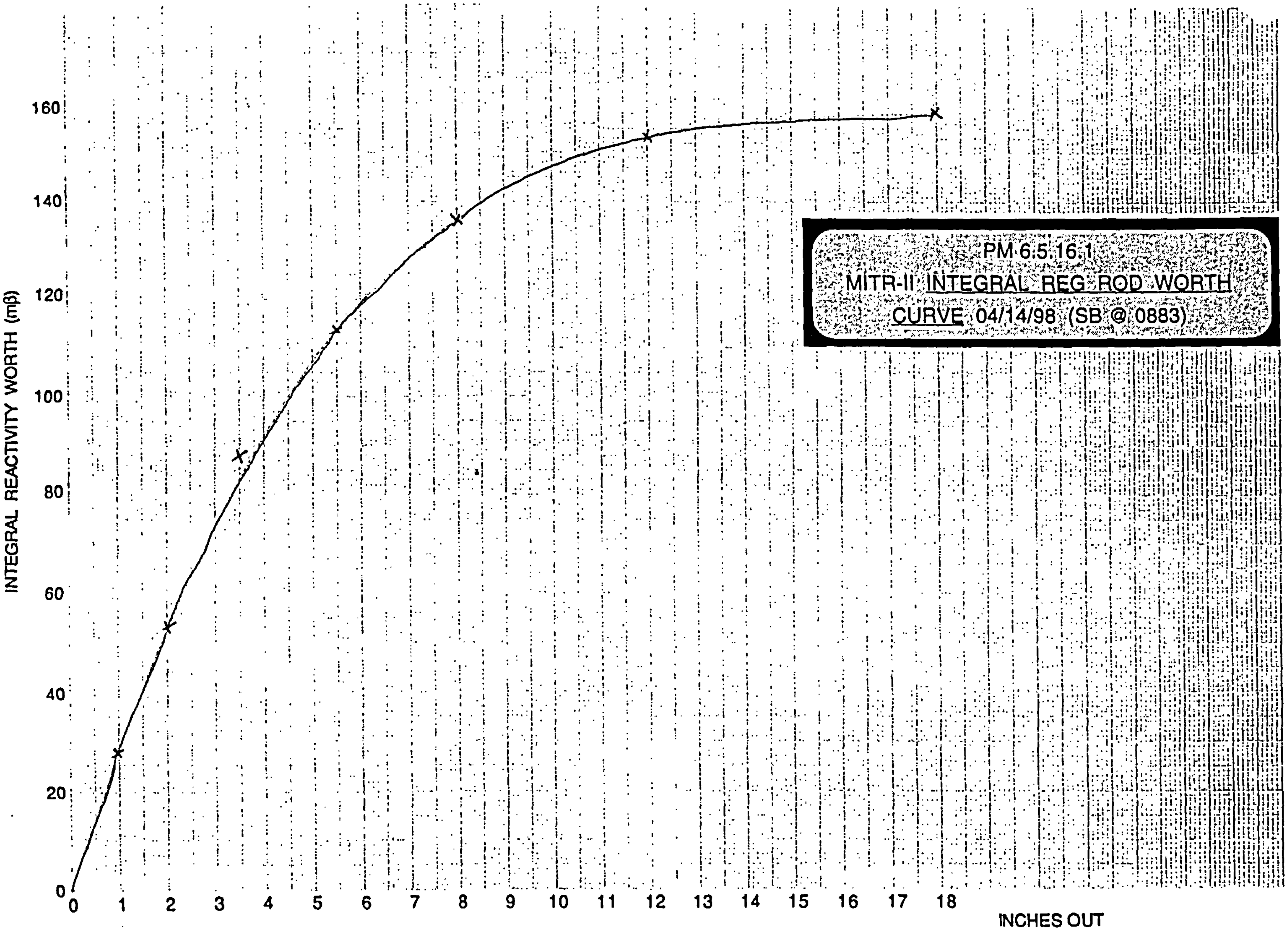


Figure 4-7 MITR Regulating Rod Integral Curve.

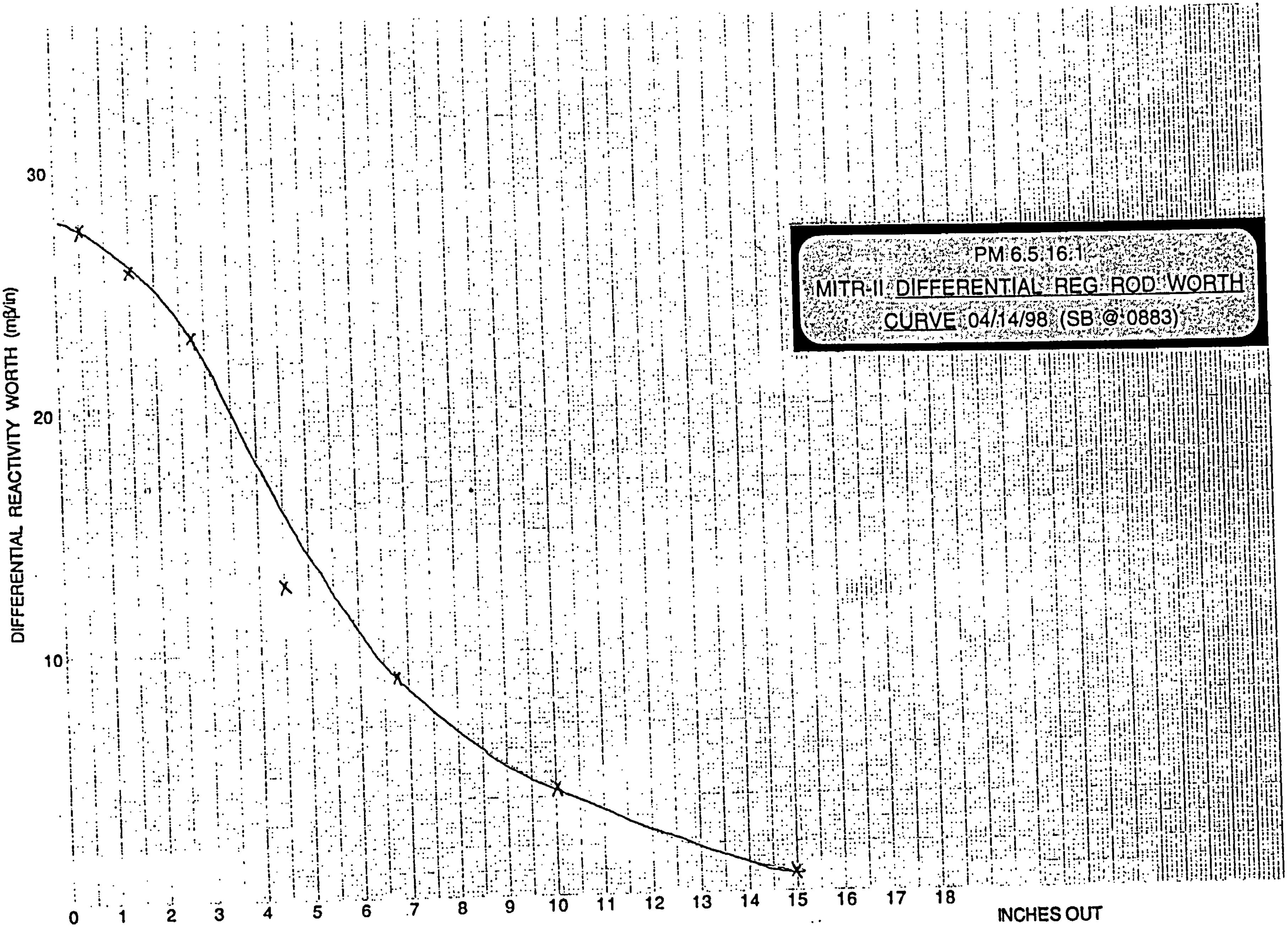
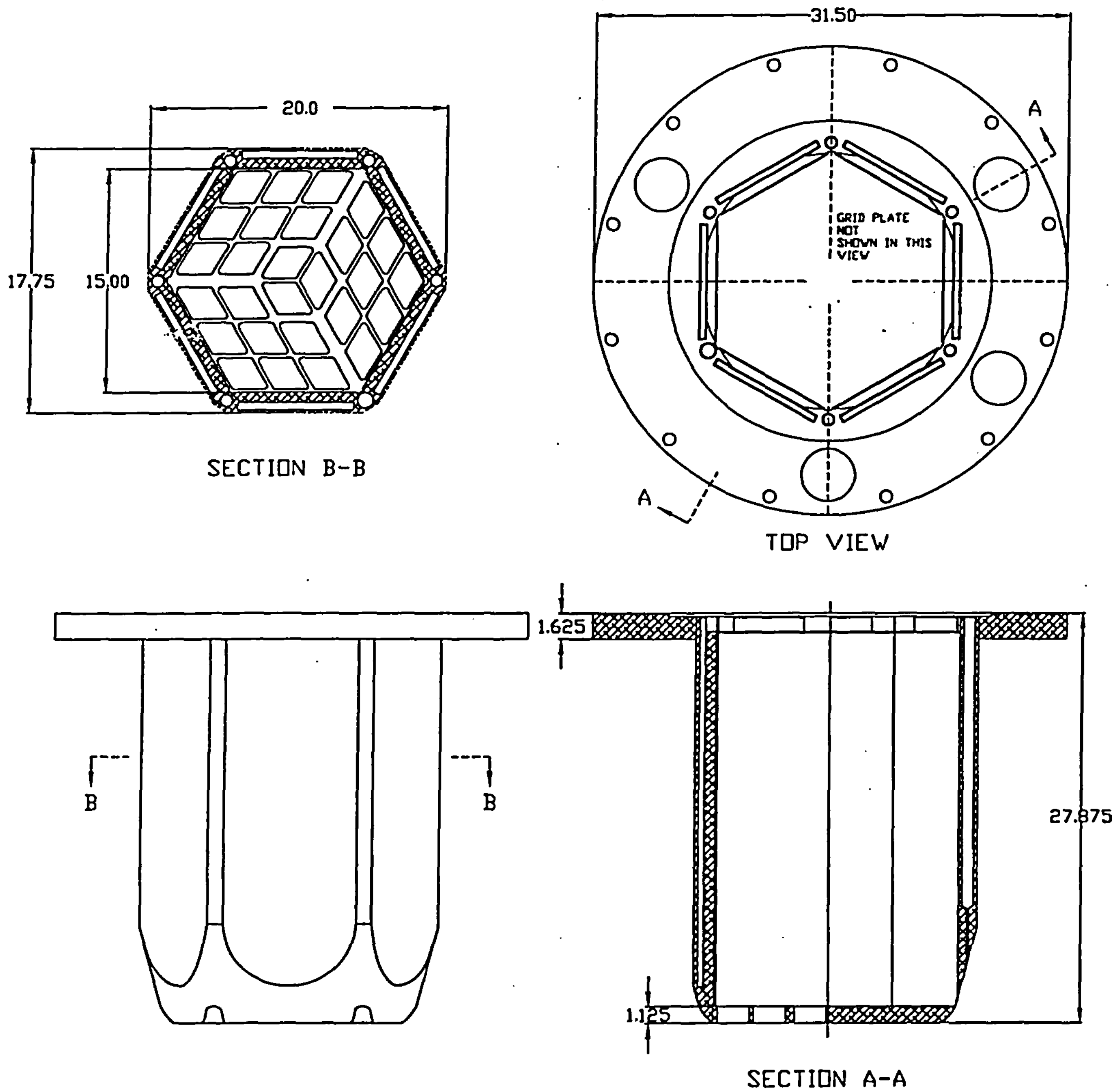


Figure 4-8 MITR Regulating Rod Differential Curve.



CORE SUPPORT HOUSING ASSEMBLY
FIGURE 4-9

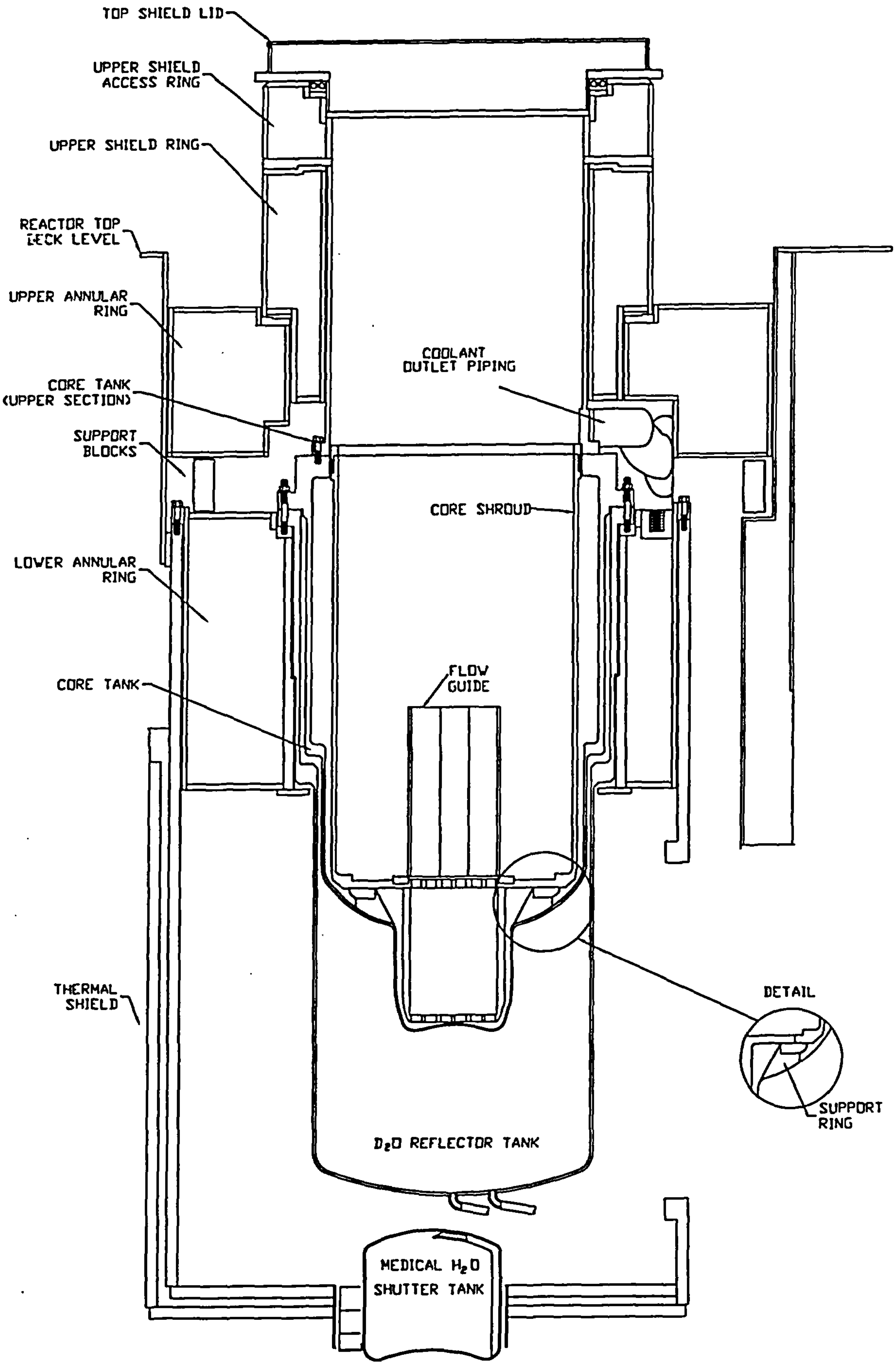


FIGURE 4-10
REACTOR CORE TANK SUPPORT

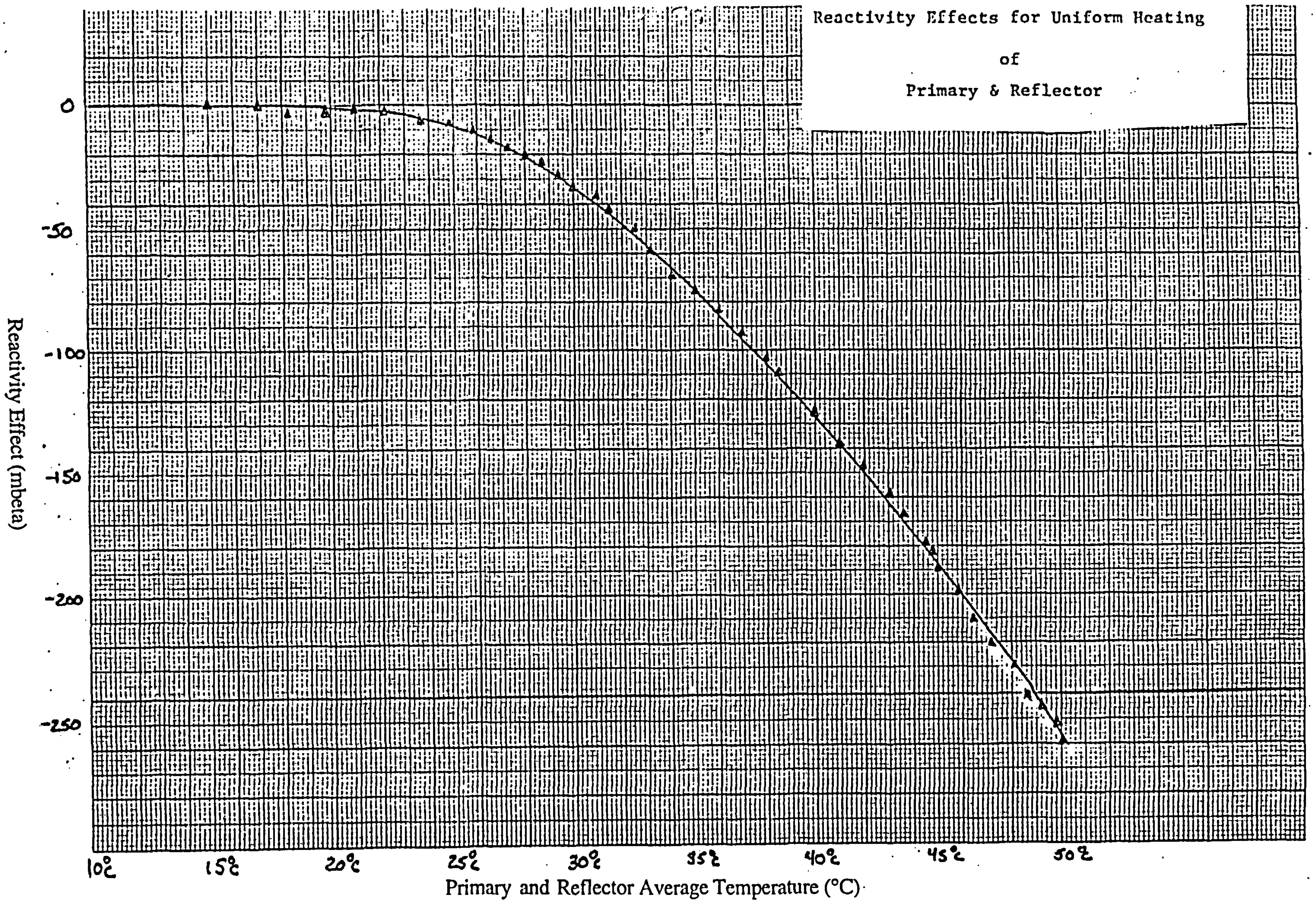


Figure 4-11 MITR Primary Coolant and Heavy Water Temperature Reactivity Effect.

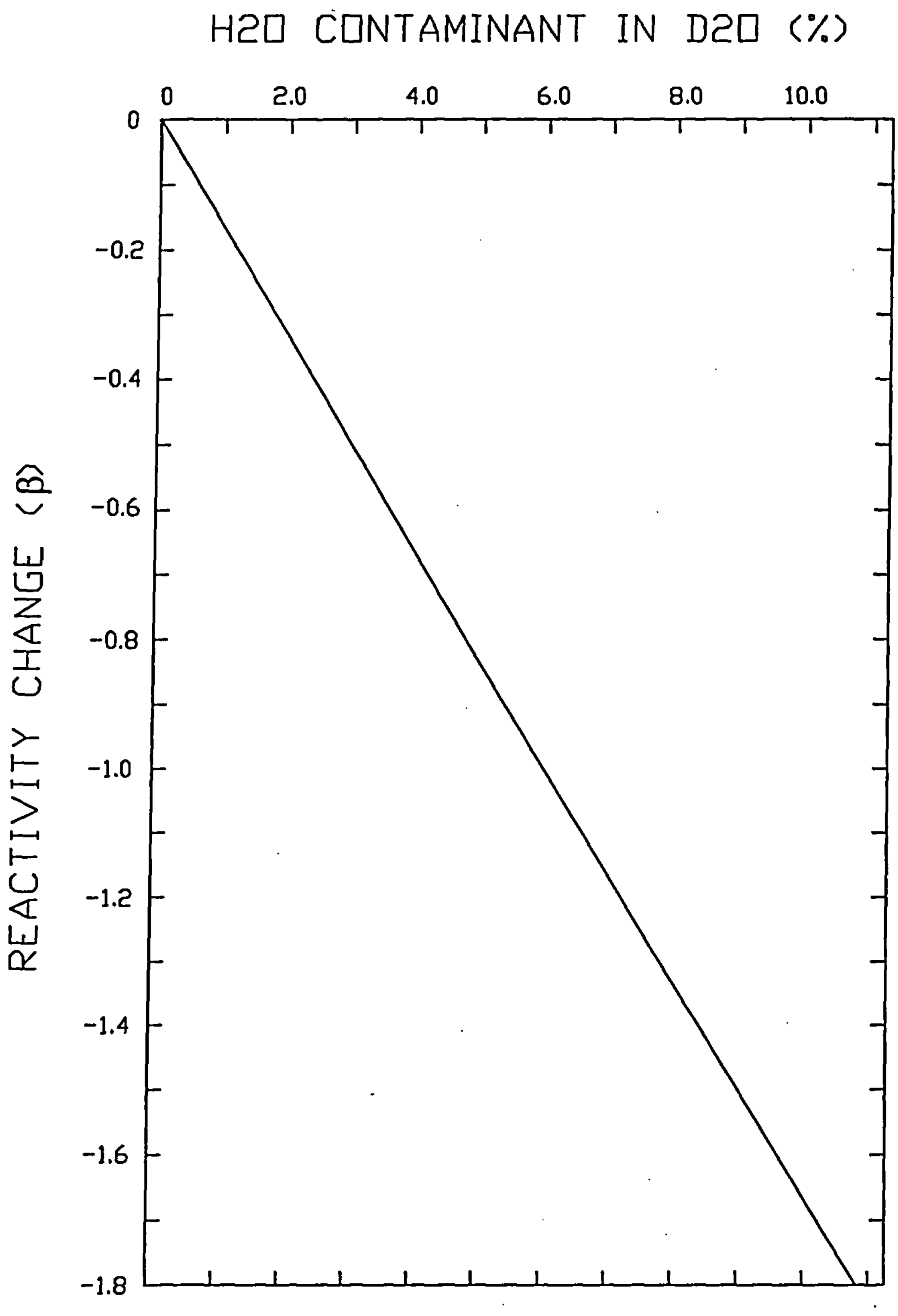


FIGURE 4-12
EFFECT OF H2O LEAKAGE INTO D2O REFLECTOR

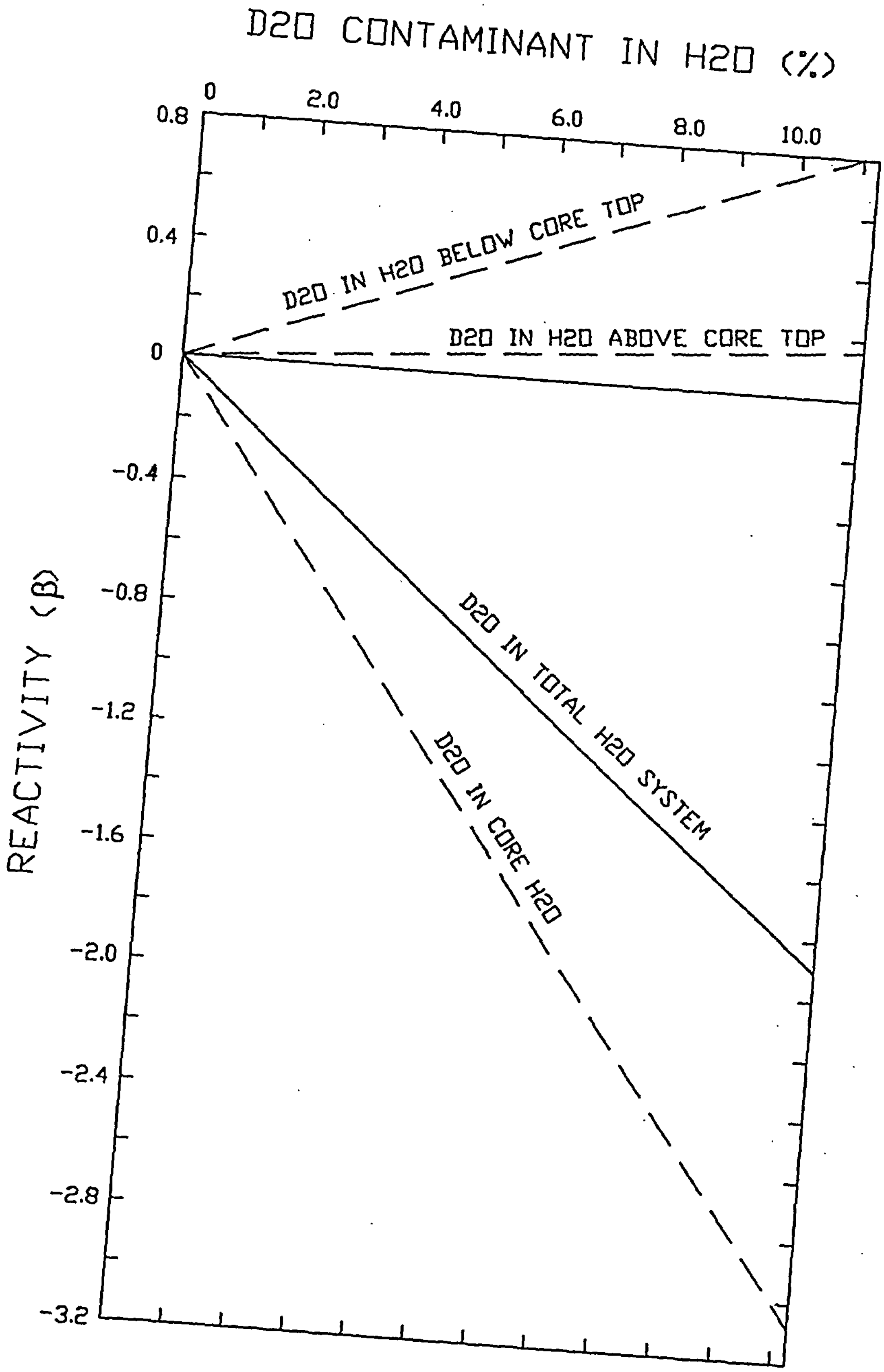


FIGURE 4-13
EFFECTS OF D2O LEAKAGE INTO H2O SYSTEM

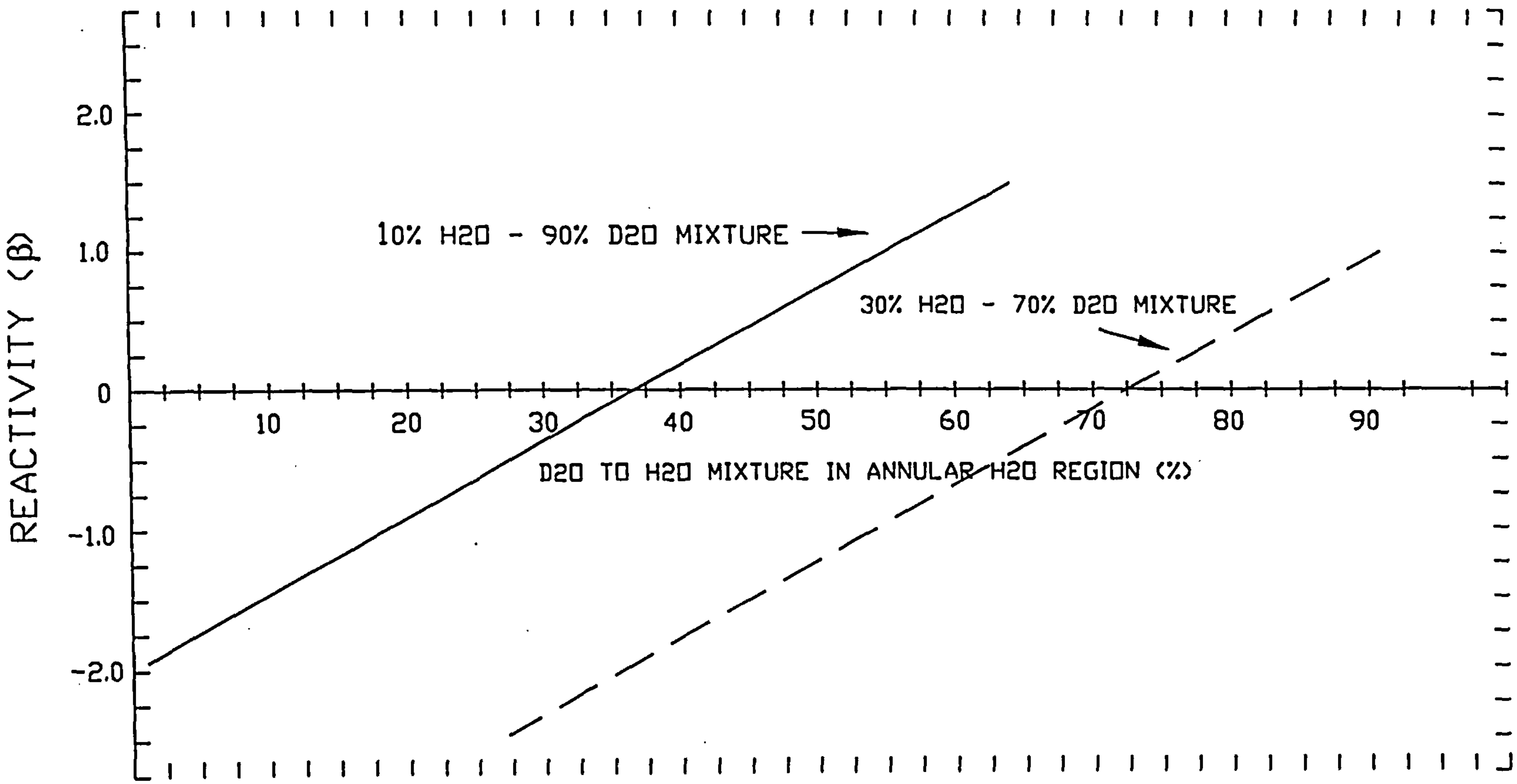


FIGURE 4-14
EFFECTS OF D2O LEAKAGE INTO H2O SYSTEM

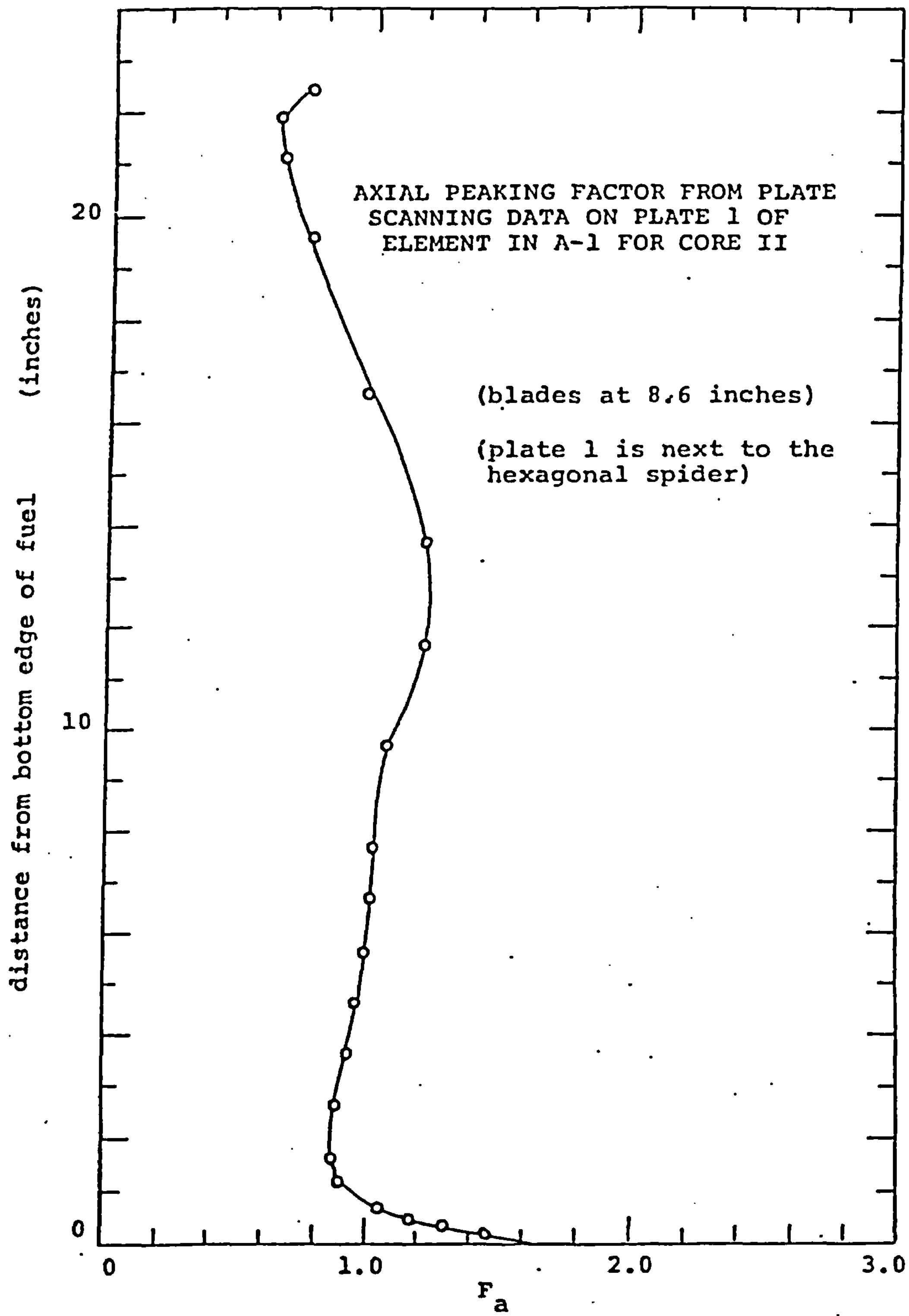


Figure 4-15 Axial Power Distribution from Plate Scanning Data on Plate 1 of Element in A-1. Adopted from [4-12].

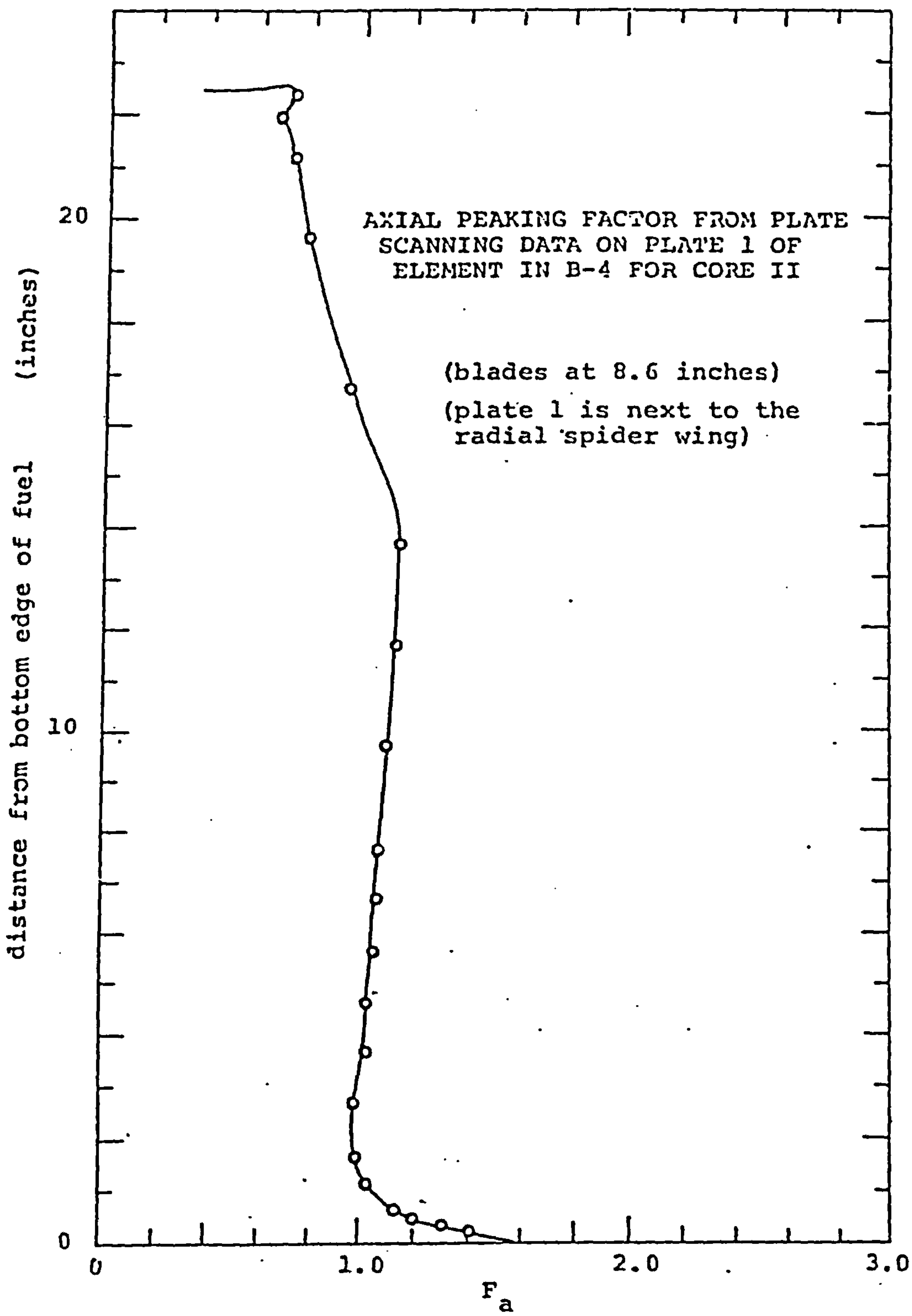


Figure 4-16 Axial Power Distribution from Plate Scanning Data on Plate 1 of Element in B-4. Adopted from [4-12].

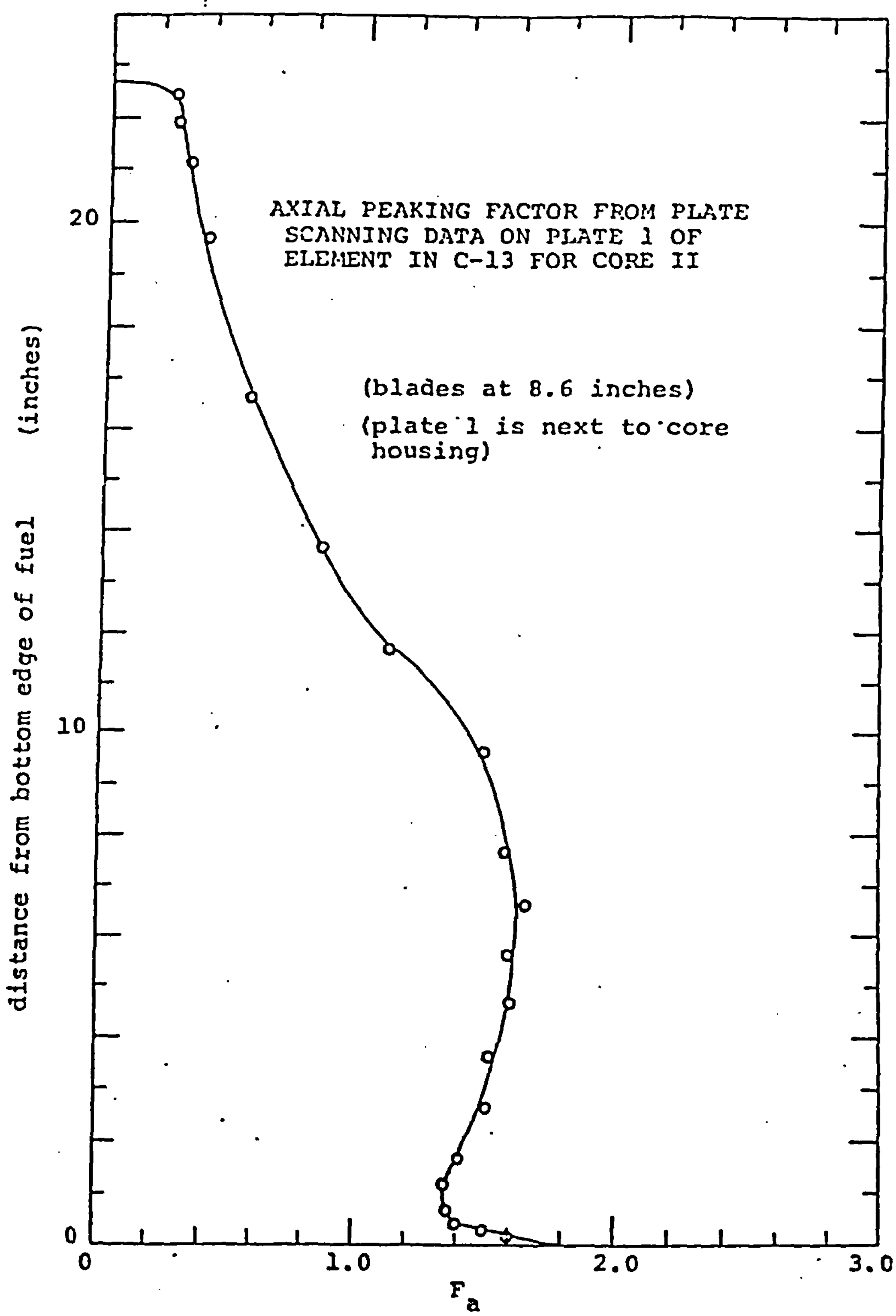


Figure 4-17 Axial Power Distribution from Plate Scanning Data on Plate 1 of Element in C-13. Adopted from [4-12].

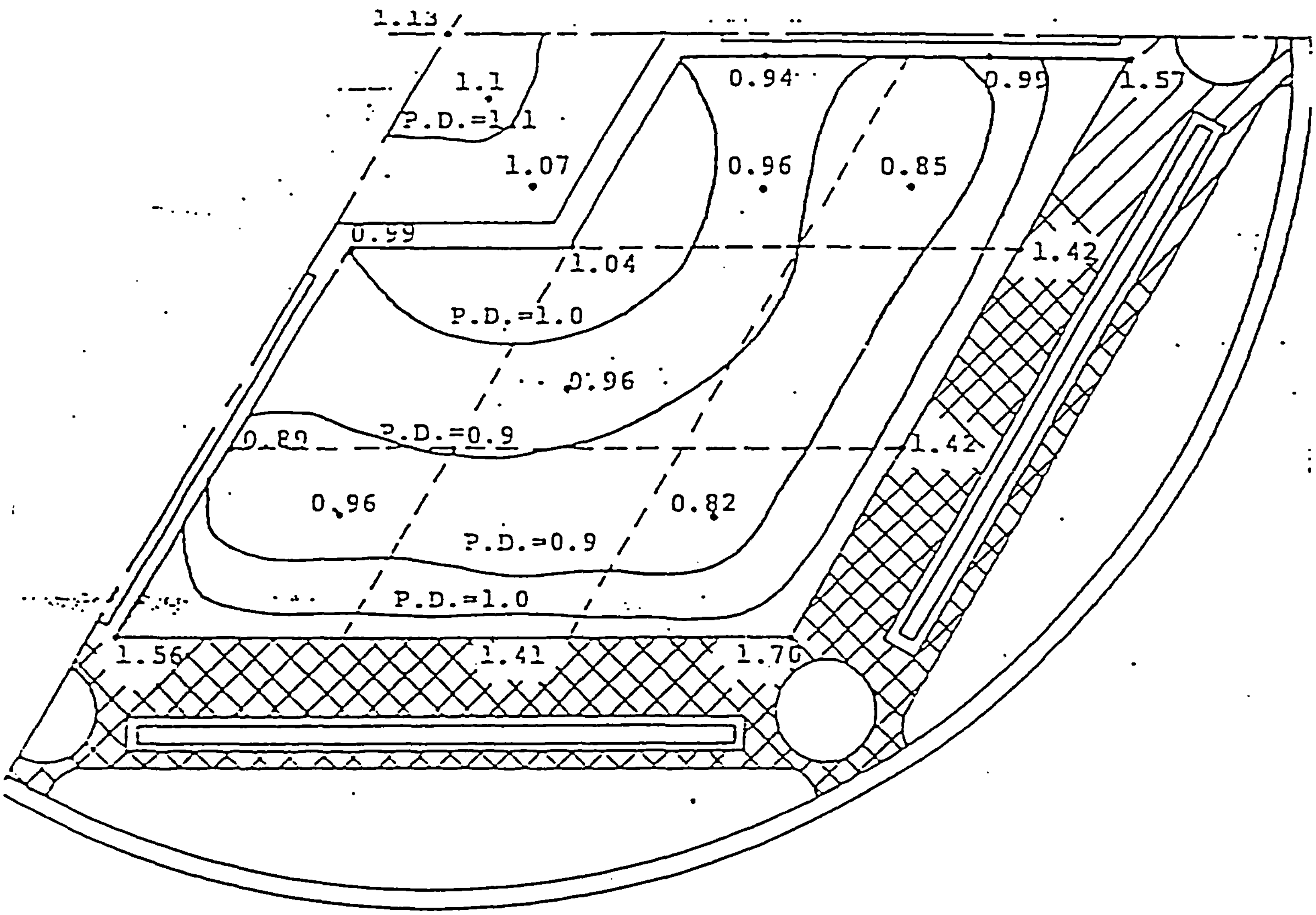


Figure 4-18 Radial Normalized Power Density Distribution in the MITR Core at Mid-Height. Adopted from [4-12].

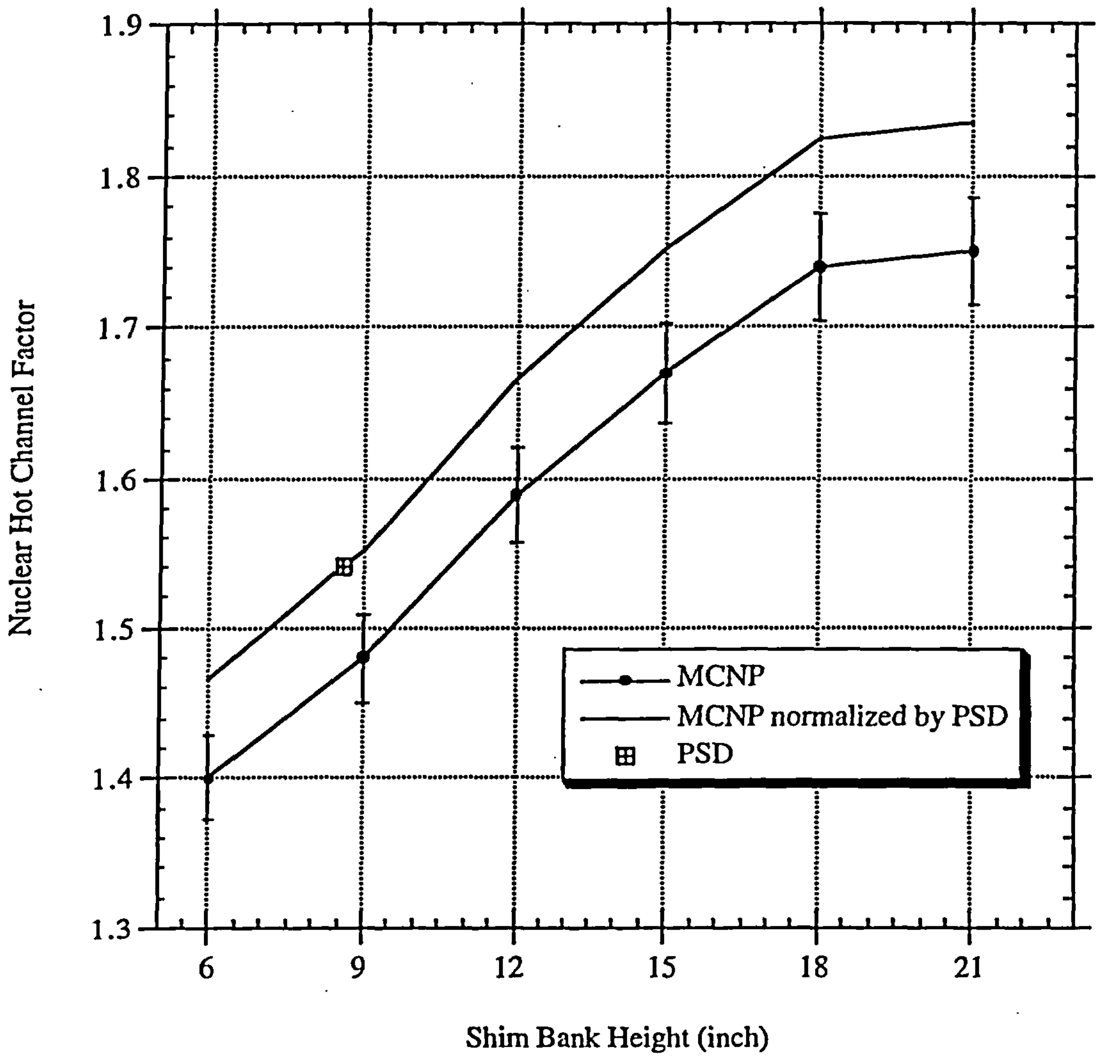


Figure 4-19 Nuclear Hot Channel Factor versus Shim Bank Height [4-24].

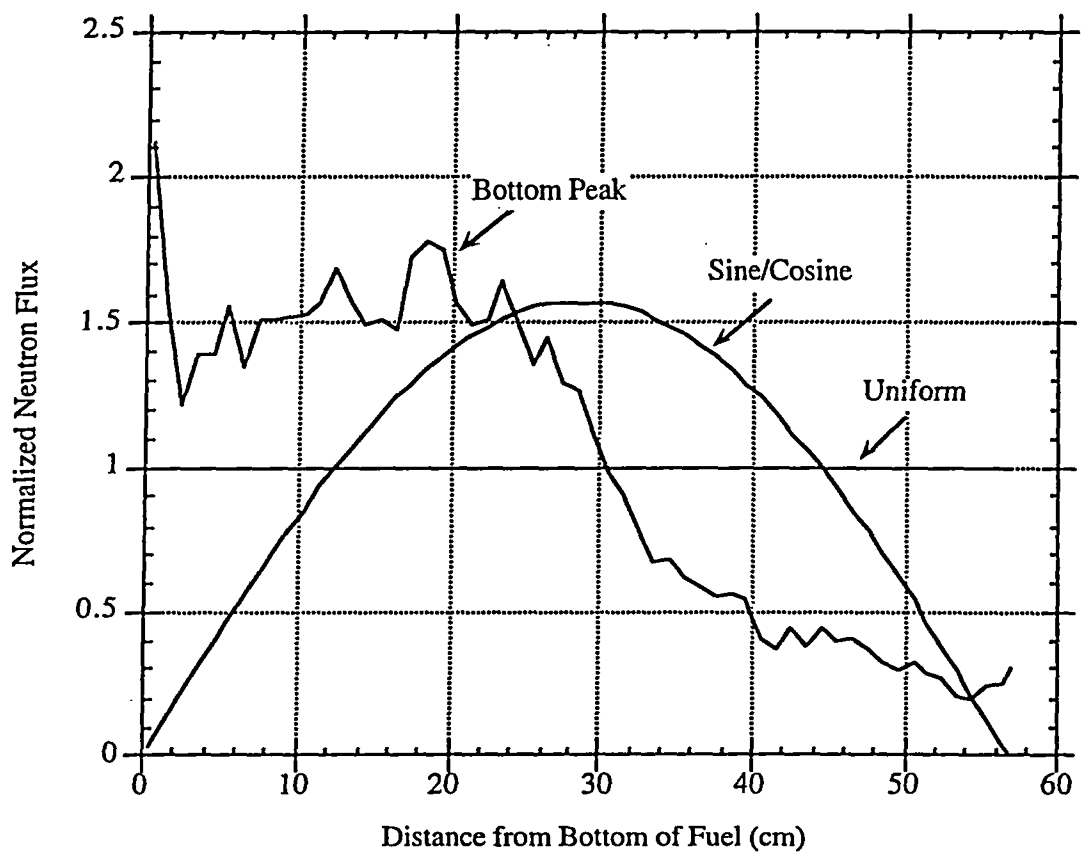


Figure 4-20 Comparison of Bottom Peak, Sine/Cosine, and Uniform Axial Power Distributions.

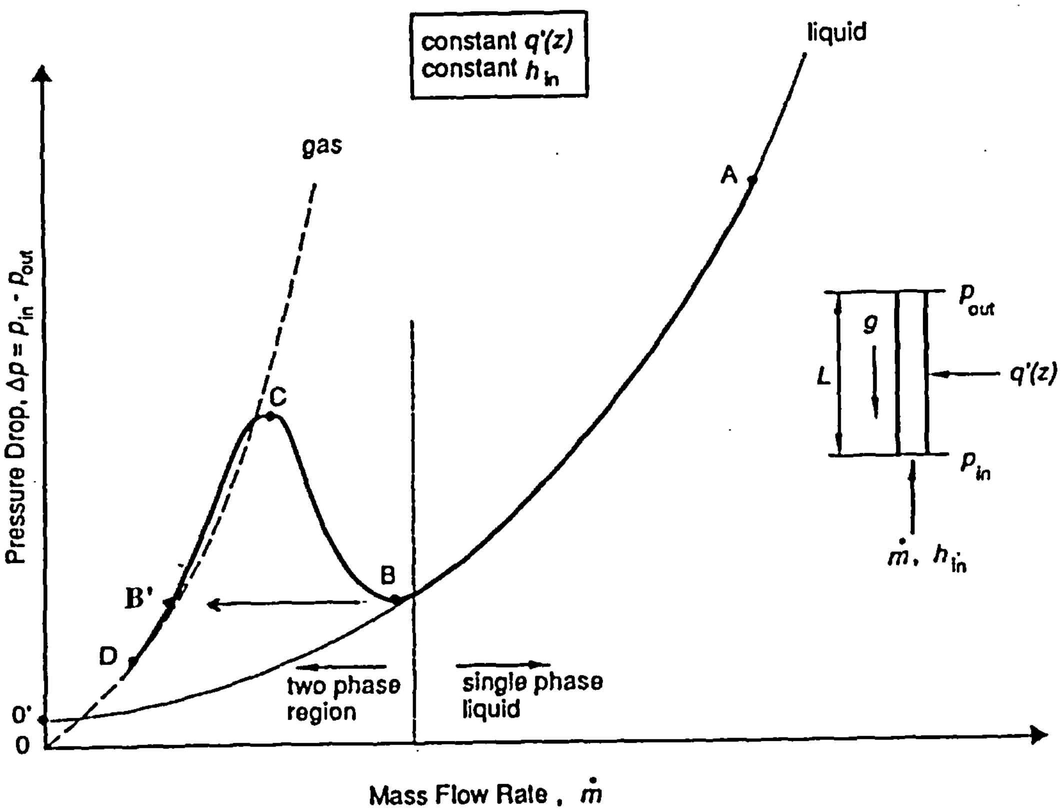


Figure 4-21 Channel Pressure Drop-Mass Flow Rate Behavior
Adopted from [4-16].

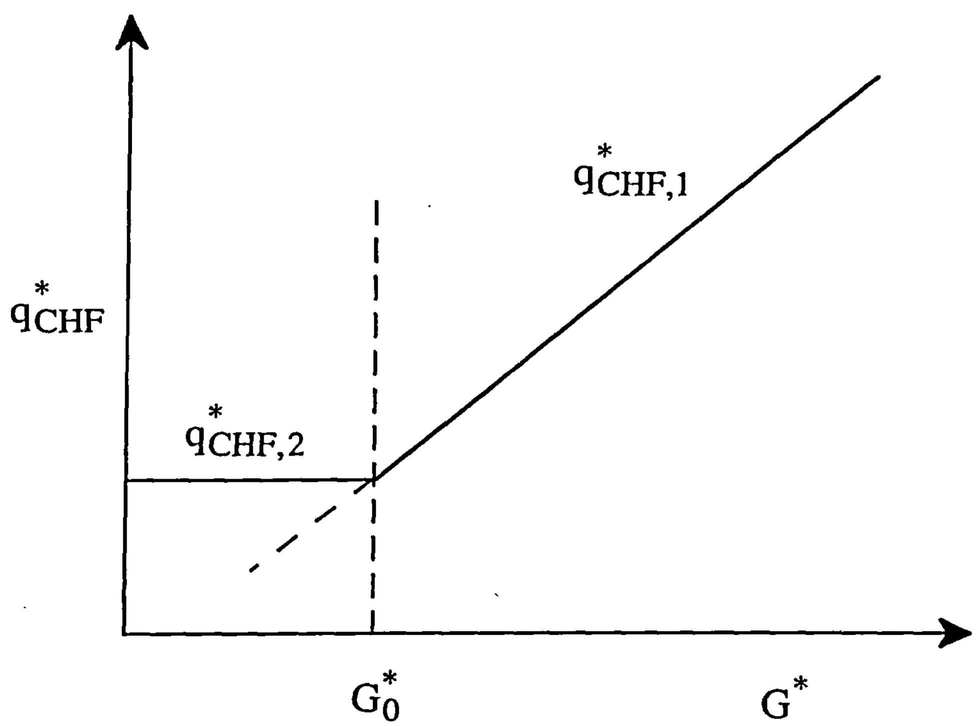


Figure 4-22 CHF Correlation Scheme for Upflow Proposed by Sudo, *et al.* Adopted from [4-20].

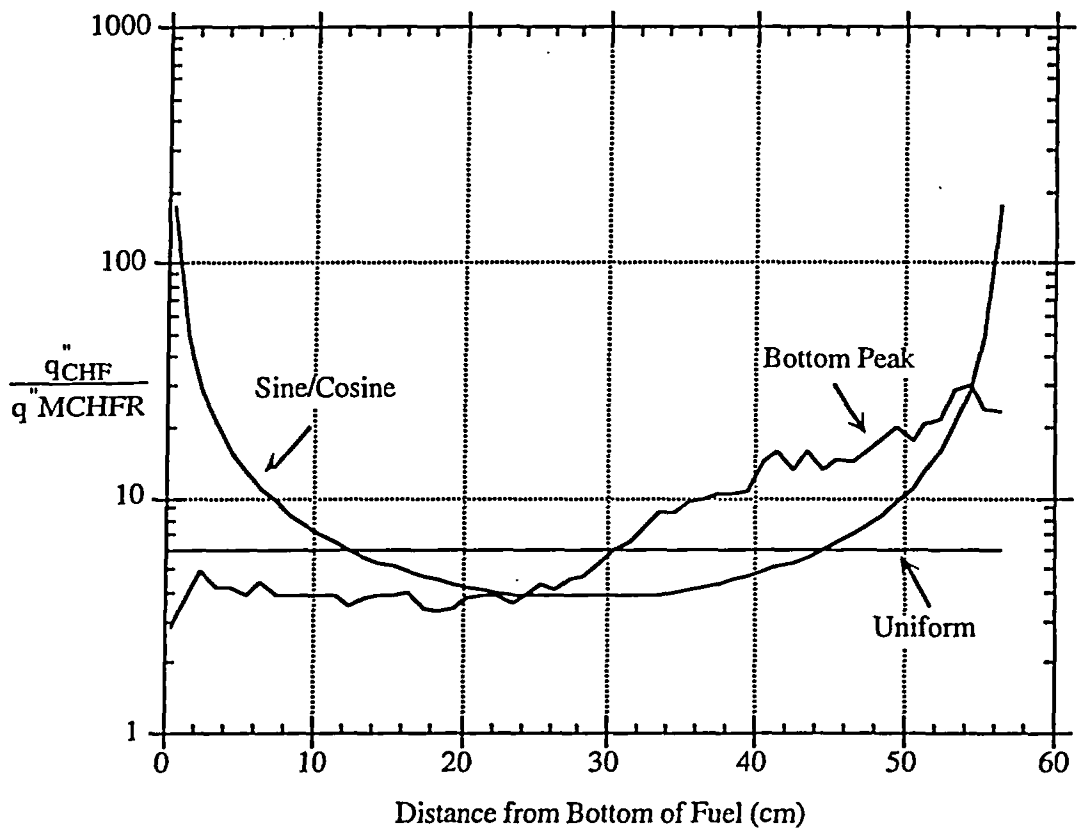


Figure 4-23 Comparison of Critical Heat Flux Ratios for Bottom Peak, Sine/Cosine, and Uniform Axial Power Distributions.

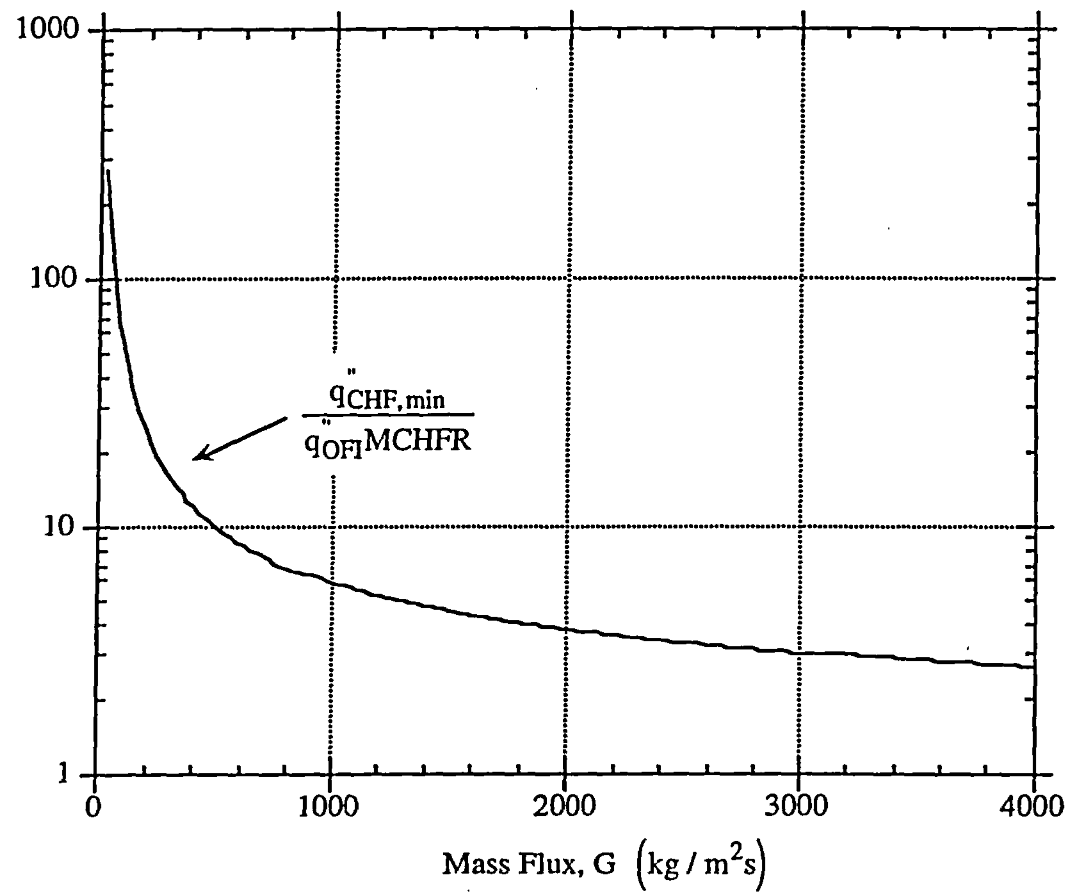


Figure 4-24 Comparison of the Minimum CHF (Obtained from Bottom-Peak Flux Distribution) And the OFI Heat Flux for Forced Flow Conditions.

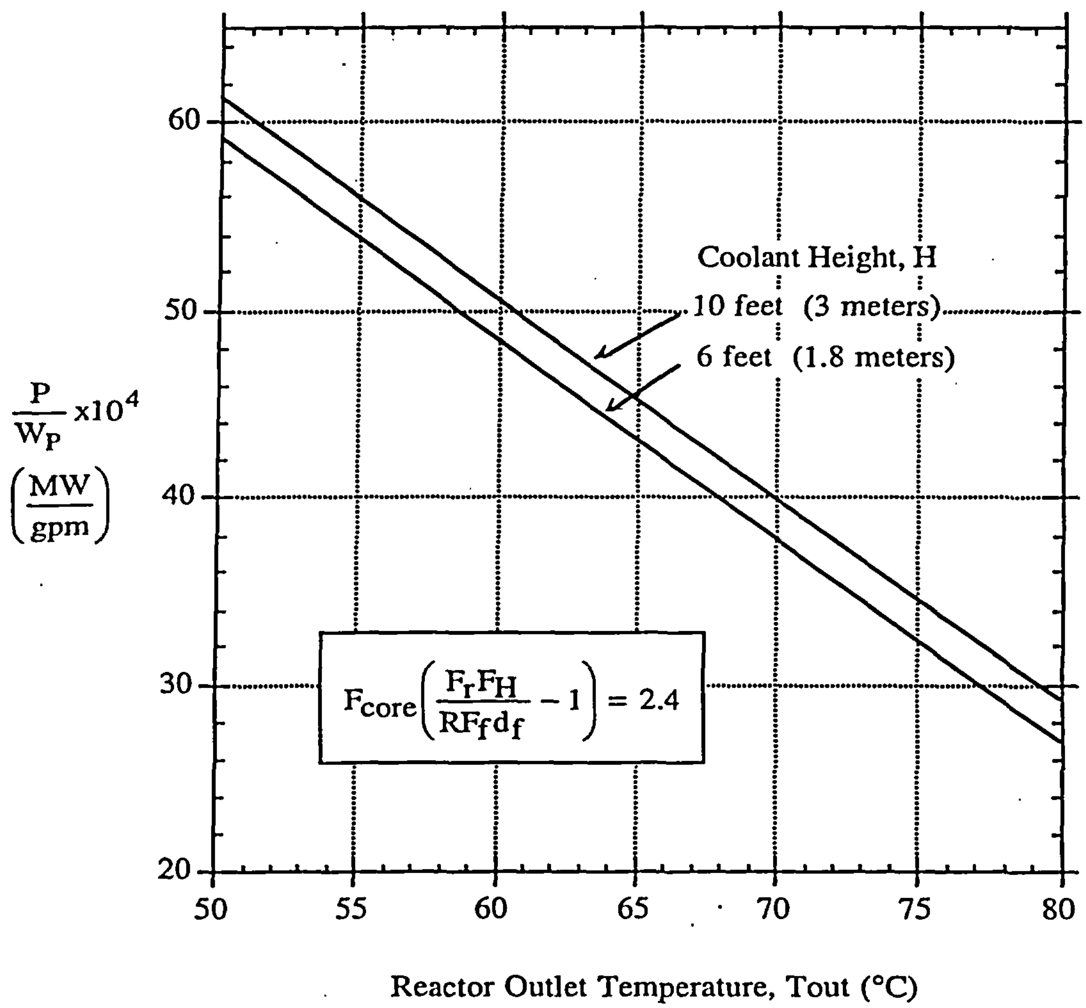


Figure 4-25 MITR-III Safety Limits for Forced-Flow Operation

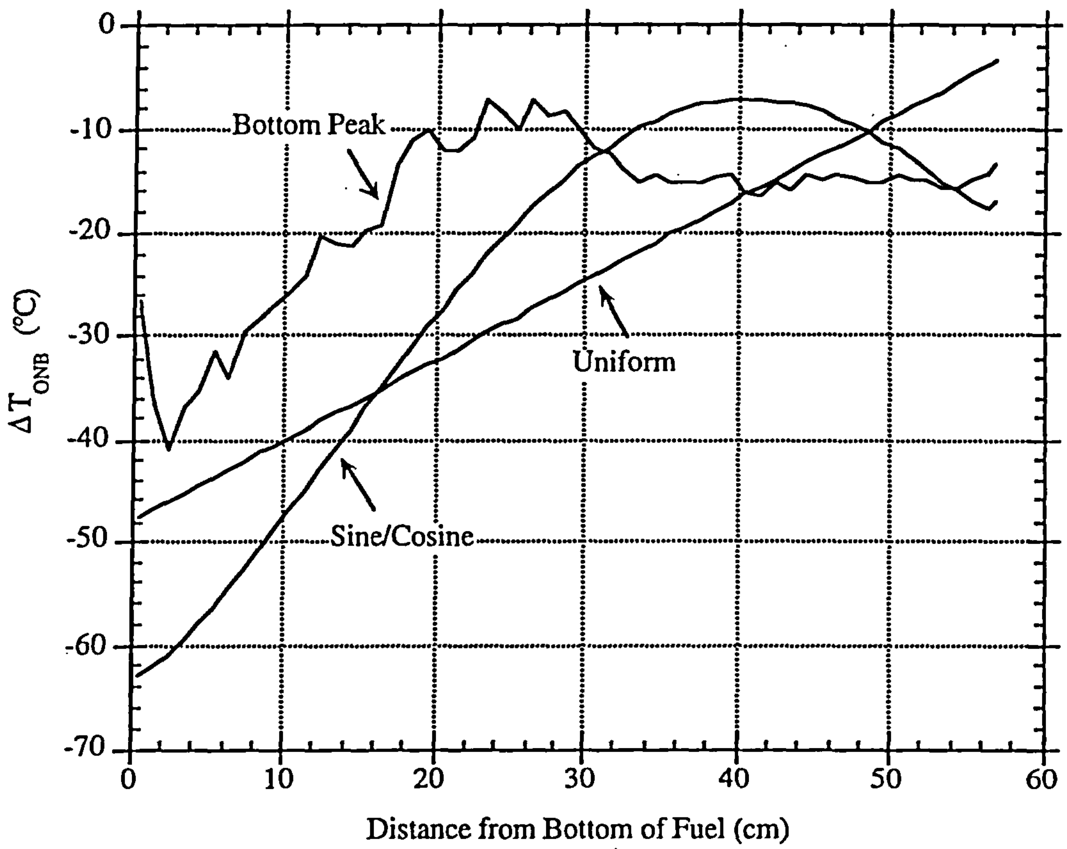


Figure 4-26 Comparison of ΔT_{ONB} for Bottom Peak, Sine/Cosine, and Uniform Axial Power Distributions.

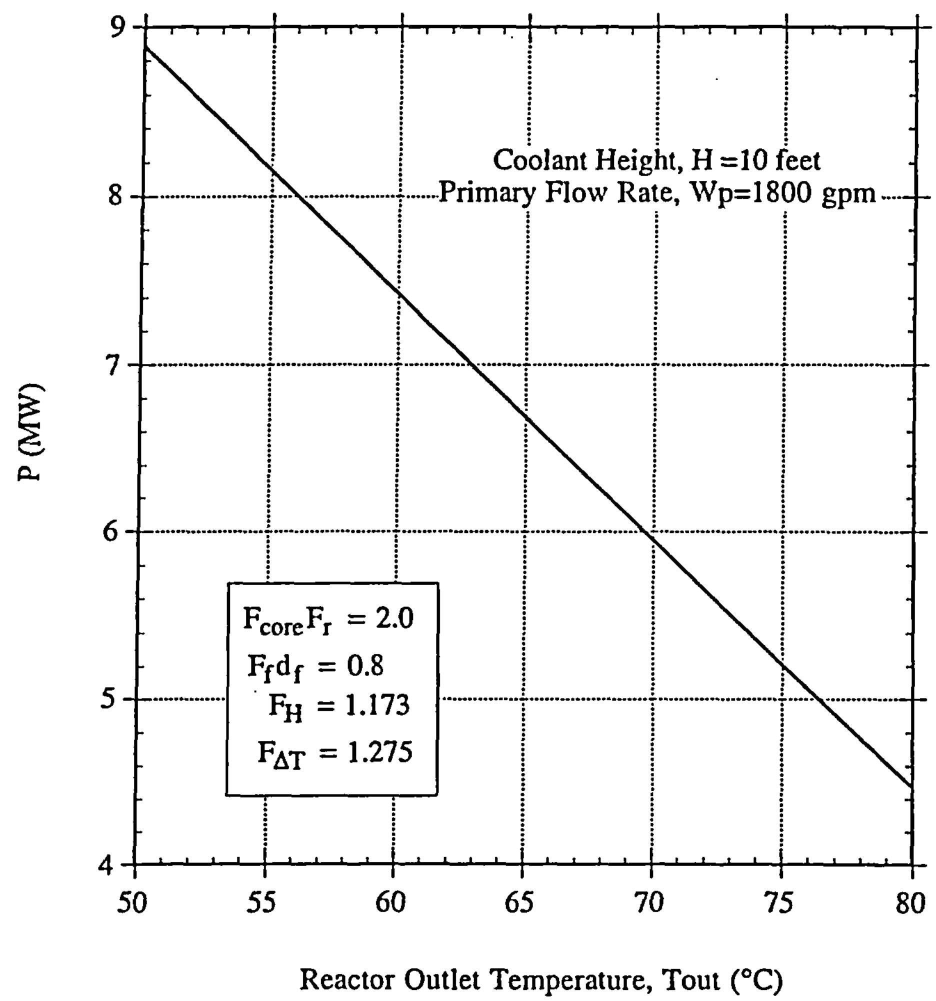


Figure 4-27 MITR-III Limiting Safety System Settings for Forced-Flow Operation (Two-Loop)

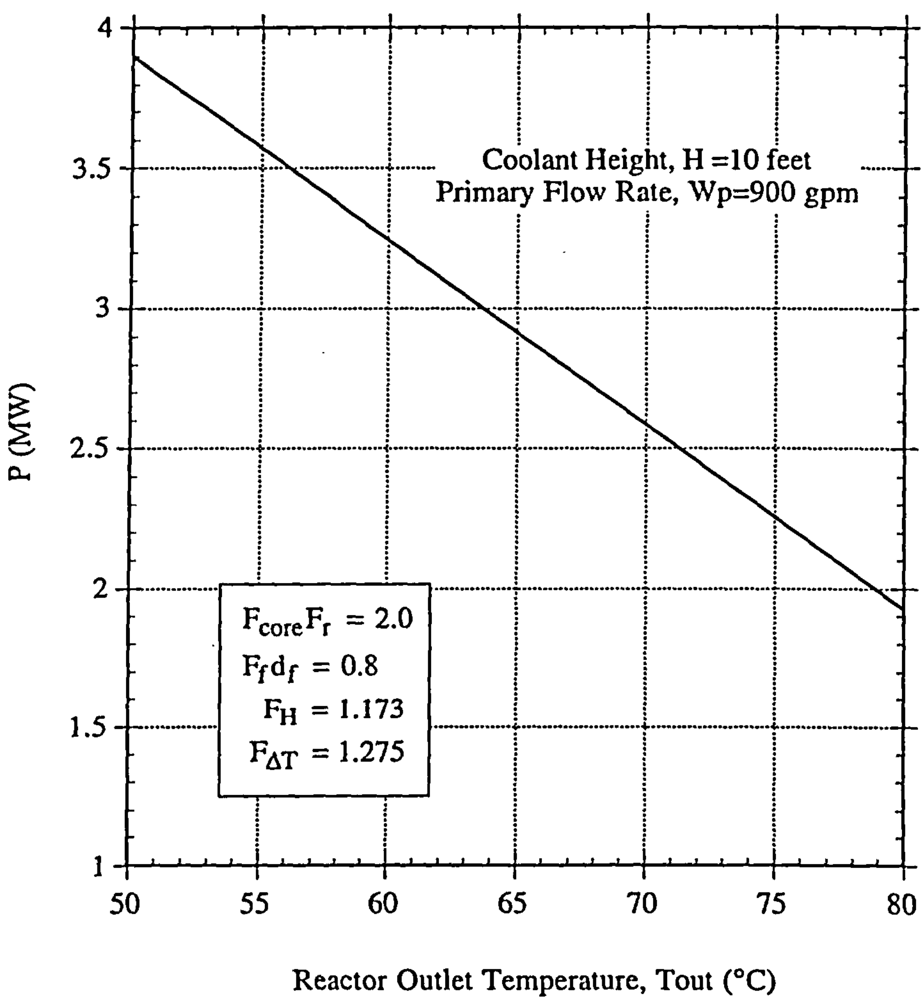


Figure 4-28 MITR-III Limiting Safety System Settings for Forced-Flow Operation (One-Loop)

Chapter 5

Reactor Coolant Systems

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Chapter 5

Reactor Coolant Systems

5.1 Summary Description

The purposes of the MIT Research Reactor's primary coolant system are to remove the heat generated in the core by fission, to moderate neutrons, to provide shielding for the reactor top, to serve as a reservoir of coolant for possible emergencies, and to act as a barrier that would retard the release of fission products in the event of multiple failures. The coolant is light water (H_2O) in the liquid state. The system, which consists of pumps and heat exchangers as well as valves and piping, is located in its entirety within the reactor containment building. A storage tank allows for the expansion and contraction of the coolant that results from temperature variations. An air purge that is drawn through the space between the surface of the coolant and the underside of the core tank's lid precludes the buildup of any radiolytic gases associated with reactor operation. The air purge system automatically isolates if abnormal radiation levels are detected. Hence, the primary coolant system is, in effect, closed to the atmosphere and it therefore serves as one of three barriers to fission product release. The other barriers, as noted in Chapters 4 and 6 of this report, are the fuel clad and the containment building. The MITR is normally operated under conditions of forced convection. However, the natural convection mode of operation is also possible at low power levels.

Under conditions of forced convection, the coolant enters the core tank through the annular space between the core tank and the core shroud and flows downward to the bottom of the core tank. It then travels upward through the core region and removes the fission energy from the fuel elements. The primary flow then mixes with the coolant in the region above the core and returns to the pumps. Heat from the core is transferred to the secondary coolant system via heat exchangers that are located downstream of the pumps.

For the natural convection mode of operation, two flow paths are involved. The first relies on natural convection to transfer hot water from the core and mix it with cooler water in the outlet plenum (the region of the core tank above the core). Natural convection valves are located above the fuel on the core support housing. These, together with the anti-siphon valves that are located in the core tank at the level of the coolant inlet pipes, promote the downward flow of coolant from the outlet plenum to the bottom of the core. The coolant then rises through the fuel and mixes within the core tank. The second flow path facilitates the removal of heat from the outlet plenum through use of the pump in the primary cleanup system loop. Energy so removed is transferred to primary cleanup system's heat exchanger, which can be cooled either by the secondary system (forced convection) or by city water that is discharged to the sanitary sewer. The cleanup system pump, which is denoted as the "auxiliary pump," can be operated using either the normal or the emergency electric power supply.

The primary cleanup loop is designed to maintain the purity of the coolant and thereby prevent excessive corrosion of the fuel elements and other materials in the primary loop. A small flow rate is drawn from the main primary coolant loop using the auxiliary pump. The cleanup flow passes through filters and an ion column to remove impurities. The cleanup system has a number of other purposes. First, as discussed above, the system may be used to remove fission energy during the natural convection mode of operation. Second, it is used to remove decay heat during reactor shutdown. Third, the system is one method for providing an emergency core cooling spray. Fourth, it provides cooling flow to the H_2O medical shutter. Fifth, it is part of the level control system for the core tank.

Beyond the core region, fission energy is deposited in the heavy water and the graphite reflectors mainly through gamma heating. This energy is removed by the reflector coolant and shield coolant systems, respectively. Each system has its own dedicated pump, heat exchanger, and cleanup system.

The secondary coolant system also uses light water. It accepts heat from the primary, reflector, and shield coolant systems, and it discharges the heat to the atmosphere via the cooling towers.

The MITR's coolant systems are capable of removing sufficient heat to support continuous full-power operation. Also, they can remove the decay heat that would be generated following extended operation at full power.

The primary coolant system has several unique features. These are the natural convection valves, the anti-siphon valves, and the spray nozzles for emergency core cooling. The purpose of the natural convection valves has already been described. These valves and the anti-siphon valves are ball-type check valves that are closed during forced convection. Both sets of valves open on loss of forced primary flow so as to promote natural convection as described above.

The purpose of the anti-siphon valves is to ensure that the core always remains submerged in coolant. Much of the primary coolant system that is exterior to the core tank is physically located below the core. Should there be a pipe break on the return portion (inlet piping), a siphon might be created. This siphon would be broken once the coolant in the core tank drops to the level of the anti-siphon valves. This feature is designed to ensure that the core itself would always remain covered.

The emergency core cooling system (ECCS) is available to spray light water onto the core. This feature should never be necessary because the core is contained in two tanks (core tank and reflector tank) and siphoning action is precluded. The ECCS can be supplied with either city water or water from the discharge of the primary cleanup loop's pump.

5.2 Primary Coolant System

5.2.1 Main Flow System

5.2.1.1 Design Bases/Functional Requirements

The primary coolant system can transfer at least 6 MW of heat from the primary to the secondary coolant system with a minimum forced flow of 1800 gpm and maintain the core free of boiling under steady-state operation. The primary coolant system fulfills several other functions:

a) The coolant pool above the core provides shielding for the reactor top.

b) The coolant pool above the core serves as a reservoir of coolant for emergency conditions.

c) The system would retard the escape of fission products and hence serves as a second barrier to fission product release (the fuel element cladding being the first and the containment building being the third).

d) The coolant is also the neutron moderator.

5.2.1.2 System Flow Path

Table 5-1

Primary Coolant System Design and Operating Parameters

Parameter	Specification or Range
Material	
Coolant	Light Water
Piping	6061 Aluminum and Stainless Steel
Flow Rate	
Minimum	1800 gpm
Nominal	2000 gpm
Temperature	
ΔT at 1800 gpm	11 °C
ΔT at 2000 gpm	10 °C
Elevation Above Top of Fuel Plates	
Overflow Line	10.3 ft
Anti-siphon Valves	6.7 ft
Inlet Penetration	6.0 ft
Construction Materials	
Core Tank/Housing	6061 Aluminum
Storage Tank	Stainless Steel
Pressure	
Pump Discharge	15 psig
Top of Core Outlet Plenum	Atmospheric
Chemistry	
pH	5.5 - 7.5
Conductivity	< 5 μS/cm
Chlorides	≤ 6 ppm
Level	
Nominal	Overflow of Core Tank
Reactor scram	4" below overflow

5.2.1.3 Heat Removal Considerations

The goal of the primary coolant system design is to remove the fission energy from the reactor core region without any boiling. A detailed description of the definition and derivation of the thermal hydraulic limits is presented in Chapter Four of this report. The requirements that are imposed on the heat removal system in order to achieve that goal are as follows:

a) Bypass flow should be minimized to allow adequate flow through the fuel element coolant channels.

b) The flow distribution in the core region should provide adequate cooling to all coolant channels.

c) The coolant height should be maintained at or above the level that was specified in the thermal-hydraulic limits calculations.

d) The primary coolant temperature at the core outlet should be kept lower than the scram setpoint to prevent boiling in the hot channel.

The above requirements apply to both forced and natural convection operation. Other relevant information is as follows:

e) Structural Features: The reactor core housing is designed to reduce bypass flow through the clearances. All the in-core positions are filled with a fuel element or an approved unit, such as a solid dummy or an in-core experiment facility, to prevent excessive bypass flow. The lower plenum of the reactor core tank has a curved shape that was designed to achieve an even flow distribution among the fuel elements. (See Figure 4-1.) The flow distribution among the fuel elements in the core, as well as the flow distribution within a fuel element, were characterized during the initial startup test of the MITR-II [5-1]. These results were adopted for the current analysis because the geometries of the core housing and fuel elements remain the same.

The upper coolant height in the core tank provides a gravity head in the core region and therefore determines the coolant saturation temperature. Because the saturation temperature increases with the pressure, the saturation temperature is reduced if the coolant height is reduced. An alarm is provided when there is no overflow in the core tank. A reactor scram is set at 4" below overflow or higher.

f) Heat Exchangers: The heat generated in the reactor core region is transferred from the primary coolant system to the secondary coolant system through the heat exchangers during forced convection. The performance of a heat exchanger is determined by its design, size, primary and secondary flow rates, and fouling. Fouling on the primary side of the heat exchangers has not been a problem because of the stringent specification on water chemistry. The heat exchangers do not impose any limit on reactor operation. The MITR is equipped with heat exchanger outlet (core inlet) temperature alarms which activate before the high core outlet temperature scram.

g) Safe Shutdown: Reactor shutdowns are normally performed with full primary coolant flow, and that flow is maintained for several hours post-shutdown to facilitate decay heat removal. Adjustments are made to the cooling towers so that the primary coolant temperature is maintained as desired during this process. However, the presence of full convection flow is not necessary for the achievement of a safe shutdown. The MITR is equipped with both natural convection and anti-siphon valves that promote the establishment of natural convective cooling. These valves, which are described in Sections 6.2 and 6.3 of this report, are a passive safety feature and thus the transition from forced to natural convection occurs automatically. The initiating event could be pump failure or a loss of off-site electricity. Regardless of the cause, if a loss of forced convective flow occurs during extended full power operation, the MITR will scram and decay heat will be removed effectively. This scenario is analyzed in Section 13.2.4 of this report.

h) Absence of Radiological Release: An uncontrolled loss or discharge of primary coolant to the environment will not occur for the following reasons. First, the primary system was well-built and it is maintained properly so that no leakage should occur. Second, locations where leaks could develop, such as pipe flanges and pump seals, are monitored for moisture. Thus, any leakage would be detected early. Third, the primary coolant system is wholly contained within the reactor containment building. If a leak did occur, the spilled coolant would be wholly contained within that building. Fourth, there is a sump in the containment. However, there is a dam around it that would prevent spilled liquid from entering the sump. Also, the sump pump is interlocked so that liquid is not pumped to the waste storage tanks if abnormal radiation levels are present. These tanks are located in the restricted area that is outside of and adjacent to the containment building. Fifth, discharge of the liquid waste storage tanks can not occur unless a locked-shut manual valve is opened and a pump started. The discharge of the pump is monitored for abnormal radiation, and the pump trips and causes an alarm in the control room if any is present.

5.2.1.4 Control and Safety Instrumentation

On/off controllers for the primary coolant pumps are located in the reactor control room. Most primary coolant system valves are positioned manually. However, those for core purge isolation and the control of the light-water medical shutter are solenoid-operated. Table 5-2 lists the safety instrumentation associated with the primary coolant system. The location of these instruments is shown in Figure 5-1. Flow, level, and temperature are all indicated in the control room. Temperature is also shown locally on the reactor top. Level is also indicated in the utility room which is outside the containment building.

5.2.1.5 Special Features

The special feature of passive decay heat removal has been described in Sections 5.1, 5.2.1.2, and 5.2.1.3 of this report. Also, a special feature to dump the heavy water reflector for backup reactor shutdown is described in Section 5.3.1.5 of this report.

Table 5-2

Primary Coolant System: Safety Instrumentation

Parameter	Designation	Function/Setpoint***
Flow	MP-6*, MP-6A*	Scram on low pressure reactor inlet at pressure corresponding to 1800 gpm or higher.
Flow	MF-1*	Scram on low flow at 1800 gpm or higher.
Level	ML-3	Scram on low level at -4 inches below overflow.
Temperature	MTS-1**, MTS-1A**	Scram on high reactor outlet temperature at 60°C or lower.
Temperature	MT-5A**	Scram on high reactor outlet temperature at 60°C or lower.

* Two out of three required.

** Two out of three required.

*** Value cited is the LSSS.

5.2.1.6 Limitation of Radiation Exposure

During normal reactor operation, primary coolant flow through the reactor core will result in the production of N-16 and Ar-41. These nuclides do not normally present a hazard to facility personnel because the primary coolant system is closed. In addition, specific design features to preclude or limit exposure are as follows:

a) The primary coolant system's pumps and heat exchangers are located in a room to which access is limited in accordance with the requirements for entry to a high radiation area.

b) The sampling system piping is of a diameter and length so that the transit time is long and hence N-16 activity decays before coolant reaches the sample station.

c) An off-gas system is used to prevent the accumulation of N-16 and Ar-41 in the space above the primary water pool. A suction on this void is drawn through a radiation monitor and the storage tank by a small blower which discharges to the main ventilation exhaust plenum. The action of this blower reduces the pressure above the pool to slightly less than atmospheric. Fresh air enters the space through an absolute filter.

5.2.1.7 Radiation Monitors

The off-gas system was discussed in the preceding section. The off-gas itself is monitored for activity using a GM pancake-type 1.4 mg/cm^2 window behind a stainless steel window. This monitor alarms in the reactor control room so as to provide early warning of abnormal radioactivity. For example, were a defect to develop in the cladding of a fuel element, this detector would sense the rise in the fission product gas concentration. Section 9.1.5.2 of this report provides additional information.

5.2.1.8 Auxiliary Systems

Auxiliary systems that use primary coolant include the primary cleanup, emergency core cooling, and light-water medical shutter systems. Each of these is shown in Figure 5-1. They are described in Sections 5.2.2, 5.2.5, and 5.2.7 of this report. The operation of these auxiliary systems has no effect on the primary coolant system's flow rate because each draws from and ultimately returns water to either the primary storage tank or another separate storage tank.

5.2.1.9 Radiation Shielding

The reactor core is contained in a tank. The arrangement of the shield blocks about this tank is shown in Figure 4-10. Whenever the reactor is operating at power levels in excess of 100 kW, the reactor top shield lid, which is a steel weldment filled with 10.38 inches of lead, is in place. If the power level is less than 100 kW, the lid may be removed. For example, this would be the situation when the reactor is shut down for maintenance. In this case, shielding is provided by the ten feet of coolant above the fuel. The design basis of the shielding is given in Section 4.4.1 of this report.

5.2.1.10 Leak Detection

The primary coolant system is equipped with water-sensitive tapes at the most probable leak areas. Alternatively, a drip tray that contains a water-sensitive probe may be placed below potentially sensitive components. The tapes and probes are connected to alarm circuits that allow the console operator to identify the point of origin of the alarm.

5.2.1.11 Coolant Radioactivity and Sampling

The primary coolant is sampled at least weekly during any week that the reactor is operating above 1 MW continuously for at least 24 hours and at least quarterly. Analyses are performed for gross activity at least weekly and for isotopic identification at least quarterly. Core performance is monitored continuously by the core purge detector. The primary coolant sample analysis serves as a backup to the indication provided by that monitor. Also, it provides a means for the detection of trends. An elevated activity level could be the result of a damaged fuel or it could be the result of an activated impurity. If the primary coolant activity exceeds 3 times the nominal fission product activity, primary coolant sampling frequency should be increased and action should be initiated to determine if any in-core fuel element is damaged and to determine if water chemistry requirements are being met. Visual inspections of the fuel elements in the core can be performed to detect possible deterioration of the clad. Also, the primary ion column will be evaluated for operability. Operation with out-of-specification chemistry is acceptable for short intervals. The important factors are pH and the absence of a high chloride ion concentration. A high conductivity by itself is not of concern.

5.2.1.12 Hydrogen Concentration Limits

H_2 gas concentration levels in the space above the primary water pool must not exceed 3.5% by volume. The minimum explosive concentration for mixtures of H_2 in the air is given as 4.1% by Sax [5-2]. A limit of 3.5% is therefore conservative.

In normal operation with a continuous purge of the air space, no buildup of radiolytic gases will occur. However, if abnormal radioactivity is detected in the purge stream, the air space above the pool will be isolated automatically. The response to such an occurrence would be for the operator to investigate the cause of the isolation and open the air space and resume the core purge as soon as possible. If core purge operation can not be resumed within fifteen minutes, reactor power will be reduced to below 100 kW. Analysis of the buildup rate in the air space above the MITR core shows that a H_2 concentration of 3.5% will not be attained during this time [5-3]. Below 100 kW, the rate of H_2O decomposition is insignificant and operation of the reactor at such power could continue with the air space isolated.

5.2.2 Primary Coolant Cleanup System

5.2.2.1 Design Bases/Functional Requirements

The principal purpose of the primary coolant cleanup system is to maintain the purity of the primary H_2O such that electrical conductivity is less than 5 μS/cm and the pH is between 5.5 and 7.5. The system fulfills other functions as well. These are:

a) Remove fission energy during natural convection operation.

b) Remove decay heat during reactor shutdown.

c) Serve as one means of providing flow to the emergency core cooling spray system.

d) Provide a cooling flow through the H_2O medical shutter tank.

e) Maintain the level of the H_2O in the core tank constant by means of continuous overflow.

f) Provide a surge volume to compensate for volume change of the H_2O because of heat up and cool down.

The purpose of pH monitoring is to ensure that corrosion of the fuel, core components, and the primary coolant loop structure is maintained within an acceptable limit. The fuel cladding and the core tank are made of aluminum alloy. A portion of the primary coolant loop is constructed of stainless steel. Lower pH will reduce aluminum alloy corrosion and oxide film formation on the fuel surface and higher pH is favored to control stainless steel corrosion. Thus, a pH range between 5.5 and 7.5 is selected for the primary coolant.

Electrical conductivity is also monitored to control purity of the primary coolant. A conductivity limit of 5 μS/cm has been traditionally adopted by research reactors.

Operation with out-of-specification chemistry is acceptable for short intervals. The important factors are pH and the absence of a high chloride ion concentration. A high conductivity by itself is not of concern. A wider pH band is acceptable for the fuel storage pool for a longer interval (one month) because the fuel is not subject to a heat flux.

5.2.2.2 System Flow Path

Figure 5-1 includes a schematic drawing and flow diagram of the primary coolant cleanup loop. It should be noted that, although the system's flow is at times part of the main flow system, the cleanup system is essentially a separate and parallel entity.

The flow path of the primary cleanup system is as follows. A suction is drawn on the storage tank, through an isolation and check valve by auxiliary pump MM-2. The pump is provided with a discharge isolation/throttle valve and a pressure gauge. A portion of the pump's discharge flows through MF-2. This flow is directed through an isolation valve to heat exchanger HE-2 where it is cooled to less than 50°C. A conductance probe, which is internally compensated for temperature, is located between the inlet filter and the ion column. A second identical conductance probe is located downstream of the outlet filter. Both filters and the ion column have individual isolation and bypass valves.

The cleanup flow returns through two isolation valves and a check valve to the suction of the main loop pumps. In the core tank, this extra water overflows through a pipe which opens 10 feet above the top of the core, passes through an overflow probe, and reenters the storage tank. Core level is therefore maintained at the height of the overflow pipe. Volume changes that are the result of temperature changes or leaks will be seen as level changes in the storage tank.

The discharge flow from pump MM-2 can be directed to locations other than the cleanup loop. Two paths are possible. First, a portion is routinely used to provide cooling to the light-water medical shutter tank. Second, flow can be directed to the emergency core cooling system. Sections 5.2.7 and 5.2.5 of this report describe these system features.

5.2.2.3 Specifications

Flow through the cleanup system is normally 2 - 5 gpm. An average value is 3 gpm. The volume of the primary coolant varies depending on the level of the storage tank. A quantity of 2000 gallons is a reasonable upper bound. Thus, the mean time to clean up the primary coolant is approximately 670 minutes. The corresponding half-life is about 7.7 hours.

The operating history of the cleanup system is excellent. Instances of out-of-specification water chemistry have been rare and of short duration.

5.2.2.4 Instrumentation

The location of the sensors and other instruments associated with the primary cleanup system is shown in Figure 5-1. The two most important instruments are the conductance probes, MC-1 and MC-2. These read out in the control room and provide an alarm if conductivity exceeds 2.0 μS/cm. Comparison of the inlet and outlet conductivity readings allows determination of the effectiveness of the cleanup system. Outlet conductivity should always be less than that of the inlet. If not, the ion exchange column is exhausted.

5.2.2.5 Schedule for Ion Column Replacement

The frequency with which the resin in an ion column requires replacement depends on the rate of introduction of impurities. For properly maintained closed systems, this rate is very low and a given charge of resin may last for several years. This is the case with the ion column for the MITR's heavy water reflector system. As for the primary system, impurities may be introduced whenever the reactor top lid is removed to allow for refuelings, experiment changeouts, training, or maintenance. These activities do not occur on a fixed schedule and hence neither does resin replacement for the primary cleanup system. A charge of resin typically lasts 8-10 weeks.

The ion column resin is not regenerated. The system is designed to incorporate two columns, only one of which is on line at a time. Once a column is identified as exhausted, it is valved off line and the spare column placed on line. The exhausted column is then removed and placed within a temporary storage space that is treated as a high radiation area. Once short-lived activities have decayed, the spent resin is discharged, dewatered, and packaged for shipment offsite.

5.2.2.6 Minimization of Exposure During Routine Operation

The ion column is located in a shielded alcove within the room that contains most of the equipment associated with the primary coolant system. That room is in turn treated as a high radiation area. This precludes any inadvertent exposure during reactor operation. Inappropriate exposures during a reactor shutdown are prevented by the performance of surveys. Specifically, the MIT Reactor Radiation Protection Office conducts surveys of the column prior to the performance of any maintenance in the vicinity of the column and before the changeout of a column. The discharge and replacement of the resin is done in accordance with an approved, written procedure.

5.2.2.7 Minimization of Exposure During Accidents

Should there be an inadvertent release of excess radioactivity in the primary coolant with a resulting deposition in the ion column, the same practices that minimize exposure during routine operation would suffice. Specifically, the ion column is kept shielded in an isolated area until surveys are performed by certified personnel, and, as necessary, revised procedure would be written, approved and issued.

5.2.2.8 Prevention of Loss of Coolant

The primary coolant cleanup system is equipped with a leak detection system as described in Section 5.3.1.10 of this report. In addition, level alarms would detect any significant leakage. Two outcomes are possible:

a) Leak rate less than MM-2 flow: In this case, the reactor core tank will remain at overflow. The first indication of the leak (other than the leak detection system) would be a low level alarm on the primary storage tank. This alarm is generated if that tank's level measurably decreases from the level at the start of operating period.

b) Leak rate greater than MM-2 flow: For this case, the core tank will not remain at overflow and an alarm will register as soon as overflow is lost.

5.2.3 Primary Coolant Sampling System

The primary coolant sampling system permits coolant samples to be taken from the main loop independent of reactor power conditions. It consists of 1/2-inch lines leading from the outlets of the heat exchangers to a sample station and then returning to the MM-1 pump suction. The sample station is mounted on a panel exterior to a biological shield wall. The pressure drop across the main coolant pumps provides sample station flow.

Analyses of the primary coolant are performed to determine gross activity, pH, conductivity, and chlorides.

5.2.4 Primary System Shutdown Cooling

Normal shutdown cooling is provided by auxiliary pump MM-2 and the cleanup loop heat exchanger HE-2. The ion column and its inlet and outlet filters may be bypassed to allow increased flow. Heat can be removed from HE-2 either by operation of the secondary cooling system or by city water that is then discharged to the sanitary sewer. The latter option would be used if off-site electricity were not available. Pump MM-2 can be powered from either normal or emergency power supplies. Hence, the capability to provide shutdown cooling does not require the availability of off-site electricity.

The decay heat generation rate is calculated to be 6.6% of reactor operating power at the instant of reactor shutdown. The decay heat decreases exponentially thereafter. This amount of heat can be easily dissipated by the shutdown cooling system. Forced convection cooling via the primary cooling system is normally maintained for three hours following a reactor shutdown after extended full-power operation. Thereafter, natural convection cooling is used.

5.2.5 Emergency Core Cooling

The MITR's emergency core cooling system (ECCS) is designed to provide protection against a loss of coolant such that the core is uncovered. This event is not a credible accident because the reactor core is contained in two concentric tanks (the light-water tank and the heavy-water reflector tank) and protection against coolant siphoning is provided by redundant valves. Nevertheless, the ECCS is installed to provide additional protection against such an event.

The ECCS is designed to provide redundant and sufficient cooling to prevent excessive fuel temperature if a loss of coolant accident were to occur. Such an accident is not considered credible for the MITR. A schematic of the piping and valves associated with the ECCS is shown in Figure 5-1. There are four modes of emergency cooling. These are summarized in Table 5-3.

Valves MV-67 and MV-68 are normally aligned to city water. ECCS Mode 3/Mode 4 can be initiated from either the reactor control room or the utility room. The latter is exterior to the reactor containment building. Thus, entry to the building, which might impose significant radiation exposure during accident conditions, is not necessary. Initiation of ECCS from either location results in full ECCS flow.

5.2.5.1 Sources of Makeup Water

Makeup water is required only for emergency cooling Modes 3 and 4 which address a large-break LOCA. The supply for Mode 3 emergency cooling is city water. The initial supply of coolant for Mode 4 is the primary storage tank, which is usually at no less than 50% capacity, or 720 gallons. This supply could be replenished by the addition of water from the MITR's makeup water system which is usually at no less than 50% of its 2000 gallon capacity. Water in both the storage tank and the makeup water tank meets the chemistry specifications for primary coolant.

Table 5-3

Modes of Emergency Core Cooling

Mode 1:	Assumptions:	1. Loss of normal electric power supply. 2. All process systems are normal except for the loss of power.
	System Lineup:	The system will be aligned as per normal shutdown cooling except that MM-2 will be supplied power from the facility's emergency power supply and HE-2 will be cooled by city water that is then discharged to the sanitary sewer.
Mode 2:	Assumptions:	1. Level in the core tank cannot be maintained at the overflow level, but it has been determined that it is not dropping below the reactor inlet penetration (~6 ft above top of the fuel plates).
	System Lineup:	The system will be aligned as per Modes 3 and 4, but these modes will not be initiated unless required. As long as the conditions assumed for Mode 2 prevail, natural convection up through the core and down through the flow shroud check valves will suffice.
Mode 3:	Assumptions:	1. Level in the core tank cannot be maintained at the level of the reactor inlet penetration (~6 ft above top of the fuel plates). 2. No source of makeup, other than city water, is immediately available.
	System Lineup:	The spray nozzle 4-way valves, MV-67 and 68, are aligned to pass water from the city supply to the spray nozzles. Spray is initiated by opening MV-70 in the Utility Room and/or MV-69 in the Control Room.
Mode 4:	Assumptions:	1. Level in the core tank cannot be maintained at the level of the reactor inlet penetration (~6 ft above top of the fuel plates). 2. A source of makeup other than city water is immediately available.
	System Lineup:	The spray nozzle 4-way valves, MV-67 and 68, are aligned to pass water from MM-2 to the spray nozzles. Spray is initiated by starting MM-2.

5.2.5.2 Instrumentation and Interlocks

The instrumentation for all modes of emergency cooling (loss of off-site power and loss of coolant) is the same. Temperature and level sensors are provided that either do not require electricity to operate or are powered from the emergency power supply. These instruments have a special feature in that, in addition to indicating locally, they can be read without entry to the reactor containment building. This feature, coupled with the aforementioned capability to initiate ECCS flow from outside the building, can minimize radiation exposures in the unlikely event of loss of coolant.

The interlocks associated with ECCS Mode 1 and 2 are those of the emergency electric supply. Vital equipment and instrumentation is automatically picked up by this power supply on a loss of off-site electricity. The system is described in Section 8.2 of this report. The interlocks associated with ECCS Mode 3 are mechanical. These consist of:

a) Backflow preventers that prevent any water from the reactor building from flowing back to the potable water supply.

b) Quick-connect couplings that must be hooked up in order to initiate ECCS Mode 3. This precludes the inadvertent introduction of city water to the core tank.

5.2.5.3 Administrative Controls

Mechanical interlocks are used to prevent leakage of contaminated coolant into the potable water supply. Administrative controls, lock wires, and warning tags are used to alert the operator to maintain the ECCS system aligned to the city water supply.

5.2.6 Nitrogen-16 Control System

By design and procedure, nitrogen-16 is precluded as a source of radiation exposure for the MITR because: (1) the primary coolant system, except for the core purge, is closed; (2) piping and heat exchangers from which N-16 photons do emanate are contained in a dedicated room to which entry is controlled as for a high radiation area; (3) coolant sampling lines are of such a diameter and length that N-16 activity has decayed prior to coolant reaching the sample station; and (4) for areas where N-16 does create a radiation hazard, such as portions of the reactor top, administrative controls are employed to limit access.

5.2.7 Light-Water Medical Shutter Tank Cooling System

Auxiliary systems that use primary coolant are listed in Section 5.2.1.8 of this report. All except the light-water medical shutter have already been described. The light water medical shutter tank cooling system is shown in Figure 5-1 and was highlighted in the discussion of the primary cleanup system. This shutter tank is one of four components used to control the delivery of radiation to the medical therapy room located below the reactor core. (Note: The other three are a D_2O shutter tank, a lead shutter, and a boral shutter.) The flow path may be summarized as follows: a portion of the coolant from the discharge of pump MM-2 flows through BF-1, two solenoid valves, and one check valve to the bottom of the shutter tank. Once the water level reaches the top of the shutter tank, the water overflows through an isolation valve, overflow probe, and temperature detector back to the storage tank. The reason for circulating the coolant is to remove energy deposited by gamma ray attenuation. The cycling of this shutter has no effect on the volumes or flows in the other portions of the reactor coolant system because coolant is drawn from and returned to the light water medical shutter storage tank.

Operational features of this shutter and associated safety considerations are as follows. The H_2O shutter tank is normally full of water, thereby preventing radiation from entering the medical room. Heat deposited by gamma rays is removed by auxiliary pump MM-2 which draws water from the primary system storage tank, circulates it through the shutter tank, and returns it to the storage tank via an overflow line. Hydrogen gas, produced by radiolytic decomposition of water in the shutter tank, is vented at least quarterly to the primary system's off-gas system by cycling the shutter. Activation of the control to void the shutter causes pneumatically-operated valves BV-21 and BV-30 to close, thereby stopping the normal circulation from the storage tank. Pump BM-1 then pumps the contents of the shutter tank to an overhead storage tank in the reactor biological shielding. A detector in this pump's suction line stops the pump upon sensing that the shutter tank is empty. A check valve downstream of the pump's discharge suction valve prevents back flow. As water is pumped out of the shutter tank, helium gas from the system's gasholder fills the tank.

The controls to close the shutter cause BV-21 to open so that water drains by gravity from the overhead storage tank to the shutter tank. BV-30 is held shut for a few minutes by a time-delay relay. This prevents MM-2 from pumping extra water into the shutter tank while it is refilling.

A helium atmosphere, supplied at constant pressure by a dedicated gasholder, blankets all free surfaces in this system. Any moisture which becomes entrained in the helium gas is collected in condensate tanks. These tanks must be drained manually.

5.3 Reflector Coolant System

5.3.1 Main Flow System

The purpose of the MITR's heavy water system is to reflect neutrons. In addition, the system provides an alternate mechanism for the rapid shutdown of the reactor. The reflector is heavy water (D_2O) in the liquid state. The system is closed to the atmosphere and is located completely within the reactor containment building. A dump tank allows for the volume expansion and contraction that results from temperature variations. All surfaces in contact with the D_2O are either stainless steel, rubber, or aluminum. All free surfaces are blanketed with helium. The helium is circulated through a recombiner to prevent the buildup of radiolytically-produced gases.

The reflector system is operated under conditions of forced convection. The purpose of this convection is to remove heat that results from the attenuation of photons in the D_2O. This heat is transferred via a heat exchanger to the secondary coolant system. A cleanup system is used to maintain the purity of the heavy water.

The reflector coolant system is capable of continuously removing the fraction of the fission energy (2-3%) that is deposited in the heavy water.

The heavy-water reflector incorporates one special feature that allows the portion of the reflector that is opposite the fuel to be rapidly drained to a dump tank, thereby introducing negative reactivity and making the reactor subcritical. This is a backup reactor shutdown mechanism.

Figure 5-2 is a schematic diagram of the reflector coolant system. The reactor core tank is contained within the reflector tank. Thus, the reflector tank is a physical backup to the reactor core tank. The space between the core and the reflector tank is filled with heavy water to a level corresponding to the upper end of the fuel elements in the core. This is where the neutron reflection occurs.

To minimize both the number of penetrations into the reflector tank and the space needed for pipe runs, one eight-inch pipe runs from the tank to the equipment room where the heat exchanger, pump, and cleanup system are located. Contained within this pipe are additional smaller pipes: one 3-inch line, one 1.5-inch line, one 0.75-inch line, and one 0.125-inch tube. These four lines emerge from the wall of the eight-inch pipe at a special elbow. The eight-inch pipe then decreases in diameter to six inches for its run through the dump valve to the dump tank.

5.3.1.1 Design Bases/Functional Requirements

The purpose of the reflector coolant system is to remove the heat deposited in the reflector by photon attenuation when the reactor is operated continuously at 6 MW. In addition, this system provides an alternate means for the rapid shutdown of the reactor.

5.3.1.2 System Flow Path

Figure 5-2 is a schematic diagram of the system. It shows the heat source (reactor core), heat sink (heat exchanger), pump, piping, valves, and control and safety instrumentation, as well as the related subsystems (recombination, cleanup, D_2O shutter, and cover gas). Table 5-4 lists design and operating parameters.

The flow path of the reflector coolant system is as follows. The main circulating pump, DM-1, takes its suction from the six-inch diameter section of the dump line before it reaches the dump valve. An isolation valve, temperature detector, and pressure gauge are located in the suction line. Temperature detector DT-1 will produce an alarm if the temperature of the heavy water coming out of the reflector tank exceeds its setpoint. An additional temperature detector, DT-5, is mounted inside the 0.125 inch tube and extends inside the special elbow to the upper part of the reflector. Its output is displayed on the multi-point temperature recorder in the control room and will cause an alarm if the temperature exceeds its setpoint.

DM-1 discharges through the tube side of heat exchanger HE-D1 to the three-inch line and then returns to the reflector tank. The flow path is therefore a closed loop that, in addition to pump DM-1, contains a pump discharge pressure gauge, pump isolation valves, a pressure gauge on the outlet of the heat exchanger, a heat exchanger isolation valve, a heat exchanger outlet temperature detector, and the flow orifice plate for DF-1. The flow detector, DF-1, provides a reactor scram if the flow of heavy water going into the reflector tank drops below its setpoint of 75 gpm or higher.

5.3.1.3 Heat Removal Considerations

The heat removed by HE-D1 is about 2 - 3% of full reactor power. Analysis has shown that loss of forced convection flow is not an immediate concern because heat would be removed from the heavy water reflector via conduction to the primary coolant [5-4]. However, boiling would eventually occur. A reactor scram, DF-1, on low D_2O flow is therefore provided. Other relevant information is that:

Table 5-4

Reflector Coolant System Design and Operating Parameters

Parameter	Specification or Range
Material	
Coolant	Heavy Water
Piping	6061 Aluminum
Flow Rate	
Minimum	75 gpm
Nominal	150 gpm
Temperature	
Maximum	55 °C
Nominal	30 - 40 °C
Pressure	
Pump Discharge	~ 25 psig
Top of Reflector	Atmospheric
Chemistry	
pH	5.5 - 7.5
Conductivity	< 5 µS/cm
Level	
Nominal	Overflow of Reflector Tank
Reactor scram	4" below overflow

a) The heat exchanger used in the D_2O system's main flow loop is oversized and does not impose any limit on MITR operation.

b) Heat removal upon shutdown, including one caused by a loss of off-site electricity, is not an issue because of the aforementioned transfer of heat from the reflector to the primary via conduction.

c) The D_2O system is, like the primary system, well-built, properly maintained, and equipped with moisture detection devices. Hence, a spill of D_2O should not occur. Moreover, if one did, it should not result in an environmental release for the reasons outlined in Section 5.2.1.3(h) of this report.

5.3.1.4 Control and Safety Instrumentation

An on/off controller for the reflector coolant pump is located in the reactor control room. Most system valves are positioned manually. Exceptions are valve DV-4 which is discussed below under special features and the solenoid-operated valves that control the heavy water shutter to the medical irradiation room. Table 5-5 lists the safety instrumentation associated with the reflector coolant system. The location of these instruments is shown in Figure 5-2.

5.3.1.5 Special Features

As shown in Figure 5-2, a dump tank is located below the reflector tank. When the pneumatically-operated dump valve DV-4 is opened, the contents of the reflector tank are "dumped" to this tank. The amount of D_2O dumped is determined by the free volume in the dump tank. The level in the dump tank is maintained such that only approximately two feet of the reflector is dumped. The negative reactivity associated with this loss of reflector is sufficient to shut the reactor down. In addition, a microswitch on the valve provides a scram when the dump valve is opened and also shuts off the transfer pump, DM-2. The dump valve is operated remotely from the reactor control room, and is also operated automatically by the major scram pushbuttons.

Table 5-5

Reflector Coolant System: Safety Instrumentation

Parameter	Designation	Function/Set point***
Flow	DF-1	Scram on low flow at 75 gpm or higher.
Level	DL-6	Scram on low level at -4 inches below overflow.

***Value cited is the LSSS.

In the event of a reflector dump, the D_2O will flood the dump tank and seek its own level in the dump tank's helium vent line which connects to the void above the reflector's normal level. The system's suction and return lines are located so that full convection can be maintained even in the "dump condition."

5.3.1.6 Radiation Exposure

The reflector coolant system does not contribute significantly to the production of N-16 and Ar-41 because the heavy water is not exposed to a flux of high energy neutrons and because the reflector is a closed system. Also, as was the case with the primary coolant system, most components of the reflector system are situated in a room to which access is restricted. Hence radiation exposure from these sources is precluded.

The principal radiation hazard associated with the heavy-water reflector is tritium, which is produced by the activation of deuterium and removed by decay. If the heavy water is initially tritium-free, the tritium concentration will rise to some equilibrium value. However, because the half-life of tritium is 12.2 years, the initial portion of this rise will be almost linear with time for a given operating power.

The presence of tritium in the heavy water necessitates that special precautions be taken to avoid exposure during both routine maintenance and unforeseen events such as a spill of heavy water. These precautions include:

a) Operation of the reflector coolant system as a closed system.

b) Quality construction of the system so as to minimize the potential for leakage.

c) Use of a leak detection system to identify any incipient leakage.

d) Stringent maintenance practices, including assurance of adequate ventilation and use of protective clothing.

e) Changeout (or dilution with clean D_2O) of the reflector to preclude a tritium concentration in excess of 5 Ci/liter. Experience with the original

MITR-I (D_2O cooled and moderated) and the MITR-II (D_2O reflector) has shown that maintenance can be performed safely on the D_2O system if the tritium concentration is less than 5 Ci/liter.

5.3.1.7 Radiation Monitors

There are no radiation monitors specific to the reflector system. Radionuclide concentrations in the heavy water are monitored by regular sampling as described in Section 5.3.1.11 of this report.

5.3.1.8 Auxiliary Systems

Auxiliary systems that use the heavy-water coolant include the reflector coolant cleanup and D_2O medical shutter systems. These are described in Sections 5.3.2 and 5.3.5 of this report.

5.3.1.9 Radiation Shielding

Shielding for the heavy-water reflector coolant system is the same as that for the primary coolant system. Sections 4.4.1 and 5.2.1.9 of this report describe these features.

5.3.1.10 Leak Detection

The heavy water reflector system and important components of the primary, shield, and other systems are equipped with water-sensitive tapes at the most probable leak areas. Alternatively, a drip tray that contains a water-sensitive probe may be placed below potentially sensitive components. The tapes and probes are connected to alarm circuits that allow the console operator to identify the point of origin of the alarm.

5.3.1.11 Coolant Radioactivity and Sampling

The reflector coolant is sampled at least quarterly for gross activity and tritium. Airborne tritium activity (or the absence thereof) is verified weekly through the use of samples collected in water bubblers. The concentration of nuclides other than tritium is normally on the order of 0.01 µCi/ml total. The tritium concentration varies over time as discussed in Section 11.1.1.5 of this report.

5.3.1.12 D_2 Concentration Limits

In a report, "Flammability of Deuterium in Oxygen-Helium Mixtures," issued by the Explosives Research Center of the Bureau of Mines [5-5], it is shown that the volume percent of D_2 needed for flammability is independent of the volume percent of O_2 from 4 to 30%. The data in this report give the flammable concentration of D_2 as 7.8 volume percent at 25°C and 7.5 volume percent at 80°C. Extrapolation of these two points by a straight line approximation indicates a flammable concentration of 6.8 volume percent at a temperature of 200°C. These results are conservative because ignition in the tests was initiated at the base of the combustion tube. Therefore, D_2 gas in the space above the heavy water surfaces must not exceed 6% of the volume.

Recombination of the disassociated D_2 and O_2 is accomplished by continuously circulating the helium from above the reflector through a catalytic recombiner. The flow through the recombiner is held at approximately two cubic feet per minute, and the recombiner operates at a temperature above 50°C as measured at the middle of the reaction chamber. A rise in temperature in the recombiner is a result of the recombination process and is positive indication that the recombiner is operating properly.

The maximum temperature in the helium system will be less then 200°C under all foreseeable circumstances. Hence, it can be concluded that combustion will not occur if the D_2 concentration is kept less then 6 volume percent.

The accuracy of the experiments described in Ref. [5-5] is best indicated by a brief discussion of the data. All experimental mixtures which were flammable for the 80°C initial temperature tests were within 0.3 - 0.5 volume percent above the 7.5% deuterium mixture limit and those which were not flammable were within a similar band below the limit. Similar results are reported for the 25°C experiment. Thus, the set of experiments was carried out in such a way as to define quite accurately the limits of the region of flammability as a function of volume percent deuterium gas with from 60 to 90% helium present and the balance oxygen. The experiments were carried out at pressures of 0.5, 1.0, and 2.0 atm and the results appear to be independent of pressure within that region. These experiments cover the range of conditions and gaseous mixtures for MITR operation.

If the recombiner operation falls outside its specified range and cannot be corrected within a fifteen minute period, the reactor power will be reduced to <100 kW. Continued reactor operation at <100 kW with the recombiner out of service is considered acceptable because the rate of D_2O decomposition at this low power level is insignificant.

5.3.2 Reflector Coolant Cleanup System

5.3.2.1 Design Bases/Functional Requirements

The principal purpose of the reflector coolant cleanup system is to maintain the purity of the heavy water such that electrical conductivity is normally less than 5 μS/cm and the pH is normally between 5.5 and 7.5. The system fulfills other functions as well. These are:

a) Maintain level of the D_2O in the reflector tank constant by means of continuous overflow.

b) Provide a surge volume to compensate for volume changes of the D_2O because of heatup and cooldown.

c) Provide capability to cycle the heavy water tank that is one of four shutters that controls beam admission to the medical irradiation room below the reactor.

5.3.2.2 System Flow Path

Figure 5-2 includes a schematic drawing and flow diagram of the reflector coolant cleanup loop. Although cleanup system flow is at times part of the main reflector flow system, the cleanup system is essentially a separate, parallel system.

The flow path of the reflector cleanup system is as follows. A suction is drawn on the dump tank portion of the dual storage and dump tank by transfer pump DM-2. A low level detector, DL-2, in the dump tank will automatically secure the pump if the tank level gets low enough to cause pump cavitation. The water flows through isolation valves for both the dump tank and pump and by a pump suction pressure gauge. The pump discharges through an isolation valve, discharge pressure gauge, and flow meter. This flow is directed through an isolation valve to heat exchanger HE-D2, where it is cooled to less than 50°C so as not to damage the heat-sensitive resin bed. It then flows through an inlet filter, a mixed-bed (D-OD) ion exchanger, and an outlet filter adjacent to the cleanup loop bypass valve DV-22 (which is normally shut). Each filter loop contains conductivity cells that are temperature compensated. Temperature detector

DT-3 monitors the outlet of the heat exchanger and will warn the console operator of possible resin damage because of a high D_2O temperature by producing an alarm at 50°C. Both filters and the ion column have individual isolation and bypass valves.

The transfer pump flow is discharged through a check valve to the suction of the main loop's pump DM-1. In the reflector tank, the extra water added from the transfer pump causes flow out through the 1.5 inch line inside the eight-inch penetration, through an isolation valve, to the sight glass and level controller. The sight glass, DL-5, is provided with isolation valves and a conductance level probe, DL-6, which will produce a reactor scram if the indicated level of heavy water in the reflector tank drops 4" below overflow. Differential pressure transmitter DL-4 is connected to the sight glass and is used to monitor the reflector tank level from the control room. The auxiliary flow overflows the level controller thereby maintaining the level in the reflector tank constant. This flow continues through an isolation valve and the overflow conductance probe, DL-9, back to the dump tank. Volume changes that result from temperature variations or leaks will be seen as level changes in the dump tank.

5.3.2.3 Specifications

Flow through the cleanup system is normally 2 - 5 gpm. An average value is 3 gpm and the volume of the reflector is normally 480 gallons. Thus, the mean time to clean up the reflector coolant is 160 minutes. The corresponding half-life is 111 minutes or about 2 hours.

The operating history of the cleanup system is excellent. Instances of out-of-specification water chemistry have not occurred.

5.3.2.4 Instrumentation

The locations of the sensors and other instruments associated with the reflector cleanup system are shown in Figure 5-2. The two most important instruments are the conductance probes, DC-1 and DC-2. These read out in the control room and provide an alarm if conductivity exceeds its setpoint. Comparison of the inlet and outlet conductivity readings allows determination of the effectiveness of the cleanup system. Outlet conductivity should always be less than that of the inlet. If not, the ion exchange column is exhausted.

5.3.2.5 Schedule for Ion Column Replacement

The heavy water reflector is a closed system that is blanketed with helium. It is rarely necessary to open the system for maintenance. Hence, a given charge of resin will last for several years. There is no set schedule for replacement of the ion column. The ion exchange resin is not regenerated. Columns from the reflector system do not pose a radiation hazard except for tritium. Hence, no special shielding requirements are needed.

5.3.2.6 Minimization of Exposure During Routine Operation.

The practices and procedures used to minimize exposure from the primary system ion column are observed for reflector system's ion column as well. However, there is little penetrating radiation associated with this column.

5.3.2.7 Minimization of Exposure During Accidents

Leakage from the heavy water reflector is of concern because of the tritium contained in the heavy water. The heavy water system is equipped with a leak detection system that will alarm to notify the operator of any moisture that may result from a leak. This system is documented in Section 5.3.1.10 of this report.

The maximum allowable tritium content in the heavy water is 5 Ci/liter as indicated in Section 5.3.1.6 of this report. An evaporation rate of 5 liters per hour is estimated for a heavy water temperature of 55°C, a relative humidity of 20%, and a spill area of 6 feet in diameter. This evaporation rate corresponds to a total volume of 120 liters, or a maximum activity of 600 Curies, in 24 hours. This is higher than would actually occur and hence conservative.

The effluent concentration (EC) for tritium concentrations in air in unrestricted areas is 1×10^{-7} μCi/ml. Using this value, a conservative dilution factor of 900, and assuming the containment building ventilation is operating, it is calculated that approximately 20 EC (3 mrem) exposure off-site would be attained over 24 hours.

5.3.2.8 Prevention of Loss of Coolant

The reflector cleanup system is equipped with a leak detection system as described in Section 5.3.1.10 of this report. In addition, level alarms would detect any significant leakage. Two outcomes are possible:

a) Leak rate less than DM-2 flow: In this case, the reflector tank will remain at overflow. The first indication of the leak (other than the leak detection system) would be a low level alarm on the dump tank. This alarm is generated if that tank's level measurably decreases from the level at the start of the operating period.

b) Leak rate greater than DM-2 flow: For this case, the reflector tank will not remain at overflow and an alarm will register as soon as overflow is lost.

5.3.3 Reflector Coolant Sampling System

The reflector coolant sampling system permits D_2O samples to be taken from the main loop at any time, although it is necessary to have the main D_2O pump running for sample station flow. The system consists of half-inch lines leading from both sides of the main flow loop to the sample station which is mounted on a panel exterior to a biological shield wall.

5.3.4 Reflector System Shutdown Cooling

Whenever the reactor is shut down, the reflector system's main pump DM-1 is secured. The system is then realigned by shutting DV-3 and DV-18 and by opening DV-19. This facilitates cleanup and cooling by ensuring that cleanup flow is directed through the reflector tank.

5.3.5 Heavy-Water Medical Shutter Tank

The heavy-water medical shutter tank is an eighteen-inch diameter, twenty-five gallon, semi-elliptical tank welded onto the bottom shell of the reflector tank. It is one of four shutters that are used to control the entry of radiation to the medical therapy room that is below the reactor core. (Note: The other three are an H_2O shutter tank, a lead shutter, and a boral shutter.) When the tank is voided, it has a negative reactivity effect of ~70 mbeta which is slight.

The tank is vented into the helium cover gas and has a single fill and drain line, which is the 0.75-inch line that is inside the eight-inch line. The tank can be drained through the remotely-operated valve DV-47 to the dump tank. This reduces the available dump volume by twenty-five gallons whenever the shutter tank is voided. The tank is refilled through remotely-operated valve DV-51 by gravity from the reflector tank level controller.

5.3.6 Heavy-Water Helium Cover Gas System

All free surfaces of the heavy-water in the reflector and reflector coolant system are blanketed with helium. The schematic diagram of this system is given in Figure 5-2. The purposes of this helium cover gas are to:

a) Prevent air with entrained H_2O moisture from entering the system, coming in contact with the D_2O, and degrading it.

b) Prevent the corrosion that would be caused by nitrous oxide formation from air in the presence of high radiation fields.

c) Provide an inert, non-radioactive vehicle to circulate the disassociated D_2 and O_2 from the reflector tank to the recombiner.

Helium flows from a high-pressure manifold located exterior to the containment building through reducers and an isolation valve to a constant pressure gasholder in the equipment room, which is where the primary and reflector coolant systems are located. The helium is then distributed to all free surfaces in the reflector system. The volume in the gasholder is replenished automatically from the manifold. The gasholder is protected from overpressure (thereby offering overpressure protection to the voids it supplies) by an oil-filled loop seal. A relief valve will vent the gasholder to the ventilation exhaust if the gasholder level becomes too high. There are provisions for manually filling or venting the gasholder.

Helium and any D_2 and O_2 from the disassociation of heavy water is circulated by a blower through a catalytic recombiner. The resultant D_2O is gravity-fed to the reflector system storage tank as it forms. Provisions exist to monitor the D_2 concentration at the inlet and outlet of the recombiner as described in Section 5.3.1.12 of this report. The D_2 concentration in the helium cover gas must not exceed 6% by volume. This condition will be met if the temperature of the middle of the recombiner exceeds 50°C and the flow rate through the recombiner is between 1.5 cfm and 8.0 cfm. If either of these conditions cannot be met, reactor power must be lowered to 100 kW or less and the system sampled for D_2 gas every hour. The D_2 concentration may not exceed 1% whenever the recombiner is out-of-commission.

5.4 Shield Coolant System

5.4.1 Main Flow System

5.4.1.1 Design Bases/Functional Requirements

The purpose of this system is to remove the heat deposited in the lead thermal shields by gamma radiation. For cooling purposes, the thermal shield is divided into four separate regions: the vertical radial shield, the bottom shield, the lower annular ring shield, and the thermal column region shield. (See Figure 5-3.) Two separate sets of coils are embedded in each of these four regions with one set normally in use while the other serves as a spare.

The shield coolant system may also be used to remove heat from experimental facilities including spectrometer magnets and the graphite vertical irradiation tubes. Each such application is reviewed prior to use.

5.4.1.2 System Flow Path

Figure 5-3 is a schematic diagram of the system. It shows the heat source (reactor shielding), heat sink (heat exchanger), pump, piping, valves, and control and safety instrumentation, as well as the related subsystem, cleanup in this instance. Table 5-6 lists design and operating parameters.

The flow path of the shield coolant system is as follows. The system uses demineralized water which is supplied from the shield coolant storage tank to the suction side of the system's pump PM-1, through an isolation valve, and a suction pressure gauge. The pump discharge is monitored for pressure and temperature as it passes through the discharge isolation valve and strainer to supply the two inlet manifolds from which the coolant is distributed to the individual shield coolant coils. The return flow from the coils collects in two outlet manifolds and passes via a single return line through the PF-1 flow orifice, an isolation valve, a temperature detector, and the primary (shell) side of heat exchanger HE-3 from which it is discharged into the storage tank. Each coolant coil is supplied with an inlet throttle valve and outlet flow meter.

5.4.1.3 Heat Removal Considerations

Operation of the reactor without cooling of the lead thermal shields might cause them to melt. Accordingly, PF-1 causes a scram if the combined flow returning from the shields drops below its setpoint. In addition, pressure switch PPS-1 will cause a reactor scram on loss of pump discharge pressure.

The heat removed by HE-3 is approximately 1.5% of full reactor power.

5.4.1.4 Control and Safety Instrumentation

An on/off controller for the shield coolant pump is located in the reactor control room. All system valves are positioned manually. Table 5-7 lists the safety instrumentation associated with the shield coolant system. The location of these instruments is shown in Figure 5-3.

Table 5-6

Shield Coolant System Design and Operating Parameters

Parameter	Specification or Range
Material	
Coolant	Light Water
Piping	Aluminum, Stainless Steel, Copper
Flow Rate	
Minimum	50 gpm
Nominal	90 gpm
Temperature	
Nominal	< 45°C
Pressure	
Pump Discharge	25 psig
Top of Storage Tank	Atmospheric
Chemistry	
pH	5 - 7.5
Conductivity	< 5 μS/cm
Chlorides	< 20 ppm
Level	
Nominal	All coils filled with storage tank level between 8 and 10 inches.

Table 5-7

Shield Coolant System: Safety Instrumentation

Parameter	Designation	Function/Setpoint***
Flow	PF-1	Scram on low flow at 50 gpm or higher.
Flow	PPS-1	Scram on low pressure shield system at pressure corresponding to 50 gpm or higher.

*** Value cited is LSSS.

5.4.1.5 Special Features

None.

5.4.1.6 Radiation Exposure

The shield coolant is not activated. It may contain some dissolved radioactive material. However, amounts are so low as to be below detectable levels. Accordingly, this system does not present a radiation hazard.

5.4.1.7 Radiation Monitors

Not applicable.

5.4.1.8 Auxiliary Systems

A small amount of the shield coolant system's flow is used to cool the thimbles in the vertical holes in the graphite reflector. The system has been used to cool magnets used in conjunction with neutron spectrometers, the vacuum pump for the fatigue cracking experiment, and the hydraulic pump for the materials testing machine. However, these experiments no longer exist.

5.4.1.9 Radiation Shielding

Not applicable.

5.4.1.10 Leak Detectors

The leak detection system described in Section 5.3.1.10 of this report is used here as well. In addition, any significant leakage would be detected by a low level alarm on the shield coolant storage tank. This alarm activates upon a measurable loss of coolant from the level at the start of the operating period.

5.4.1.11 Coolant Radioactivity and Sampling

The shield coolant is sampled quarterly for gross activity. The radioactivity level is normally less than detectable levels.

5.4.1.12 Hydrogen Concentration Limits

The production of H_2 from radiolytic decomposition has never been detected in the shield system and would not be expected to exist. In any event, the shield storage tank is vented to the atmosphere, so any H_2 would dissipate.

5.4.2 Shield Coolant Cleanup System

5.4.2.1 Design Bases/Functional Requirements

The purpose of the shield coolant cleanup system is to maintain the purity of the shield coolant such that electrical conductivity is normally less than 5 μS/cm and the pH is normally between 5 and 7.5.

5.4.2.2 System Flow Path

A small portion of the discharge flow from pump PM-1 is directed through an ion column. This flow then rejoins the main flow path as shield coolant enters heat exchanger HE-3. The cleanup loop is equipped with a flow detection device, PF-10, and a conductance probe, PC-1.

5.4.2.3 Specifications

Flow through the cleanup system is normally 2 - 3 gpm. An average value is 3 gpm. The volume of the shield coolant is normally 130 gallons. Thus, the mean time to clean up the shield coolant is 43 minutes. The corresponding half-life is about 30 minutes. The operating history of the cleanup system is excellent. Instances of out-of-specification water chemistry have not occurred.

5.4.2.4 Instrumentation

The location of the sensors and other instruments associated with the shield cleanup system is shown in Figure 5-3. The most important instrument is the conductance probe PC-1, which reads out remotely.

5.4.2.5 Schedule for Ion Column Replacement

The shield system is a closed system except that overpressure protection is provided by venting the storage tank to atmosphere. It is rarely necessary to open the system for maintenance. Hence, a given charge of resin will last for several years. There is no set schedule for replacement of the ion column. The ion exchange resin is not regenerated. Columns from the shield system do not pose a radiation hazard. Hence, no special shielding requirements are needed.

5.4.2.6 Minimization of Exposure During Routine Operation.

This section is not applicable because the shield system is not a source of radiation.

5.4.2.7 Minimization of Exposure During Accidents

This section is not applicable because the shield system is not a source of radiation.

5.4.2.8 Prevention of Loss of Coolant

Section 5.4.1.10 of this report describes leak detection for this system.

5.4.3 Shield Coolant Sampling System

Shield coolant samples can be taken any time that the shield coolant pump is running. A quarterly analysis is made of the shield coolant for gross activity, pH, conductivity, and chlorides.

5.4.4 Shield System Shutdown Cooling

The shield coolant system operates continuously regardless of whether the reactor is at power or shut down. The only change made on shutdown is to reduce the amount of secondary flow to the shield heat exchanger so as to limit the rate of cooldown of the shields and thereby preclude any possibility of thermal shock.

5.5 Secondary Coolant System

5.5.1 Main Flow System

5.5.1.1 Design Bases/Functional Requirements

The secondary coolant system is designed to transfer the heat from both the reactor coolant systems (primary, reflector, shield, and experiments) and the fission converter facility to atmosphere. It is designed for continuous full-power operation and it is not shared with any other facility. At present, there are eight heat exchangers that reject heat to the secondary: three primary (HE-1, HE-1A, and HE-1B), one primary cleanup (HE-2), one reflector (HE-D1), one reflector cleanup (HE-D2), one shield (HE-3) and one for heat removal from the ex-core testing of any planned in-core experiments (HE-4). A ninth heat exchanger, that for the fission converter, is slated to be installed during 1999. The rating of the existing eight heat exchangers is at least 6.7 MW and that for the planned fission converter unit is 250 kW. Thus, the total capacity of heat exchangers is about 7 MW. The heat sink is the cooling towers which can reject 10 MW to the atmosphere under the most adverse (high humidity and temperature) weather conditions. It can reject more heat under less stringent atmospheric conditions.

5.5.1.2 System Flow Path

Figure 5-4 is a schematic diagram of the system. It shows the heat sources (eight heat exchangers), heat sink (cooling towers), pumps, piping, valves, and control and safety instrumentation. The system operates only under the condition of forced circulation.

The secondary system is normally operated in a split configuration with HV-27 closed so that HE-1 is supplied by pump HM-1 while HE-1A is supplied by pump HM-1A. HE-1B can be supplied from either pump depending on the position of the cross-connect valves. The usual lineup is described below.

Water is drawn from the cooling tower basins through individual isolation valves and a common supply header by the main secondary pumps, HM-1 and HM-1A. A resistance temperature detector, HT-1, mounted in this return line, alarms on low cooling tower outlet temperature at 10 °C to prevent possible freezing of the D_2O in the D_2O heat exchangers at 4 °C.

Water from HM-1 flows through the secondary sides of heat exchangers HE-1, HE-D1, and HE-D2 and into the secondary return header. HE-1B could be supplied by this pump or HM-1A if desired. The flow passes through an isolation valve, a check valve, the HF-1 flow orifice, and individual isolation valves for the inlet and outlet of the heat exchangers. Pressure gauges monitor the inlet and discharge pressure of HE-1 and HE-1B. The inlet pressures have remote indication in the control room. HF-1 provides a scram if the flow of secondary coolant through it drops below its setpoint.

The flow through HE-D1 and HE-D2 comes from the pump through flow indicator HF-6. Both heat exchangers have individual inlet and outlet isolation valves. HE-D1 has inlet and outlet pressure gauges. HE-D2 has an inlet pressure gauge.

Water from HM-1A flows through the secondary sides of heat exchangers HE-1A, HE-1B, and HE-3 and into the return header. Pressure gauges monitor the inlet and discharge pressures of the heat exchangers. The inlet pressures have remote indication in the control room. HF-1A provides a scram if secondary flow drops below its setpoint.

The flow through HE-3 comes from HM-1A through flow indicator HF-4. This heat exchanger has individual inlet and outlet isolation valves, an inlet pressure gauge, and an outlet temperature detector.

Heat exchanger HE-2 can be supplied from either main pump.

The secondary return header penetrates the containment shell and goes to the cooling towers. The secondary return line splits into two parallel paths with flow through each path being determined by individual isolation valves. The first of these paths allows the water to pass directly into the basin. The second allows the water to pass directly to the spray nozzles at the top of the cooling towers. Flow through these paths may be throttled if necessary to adjust cooling tower heat removal capability.

The water from the spray nozzles cascades down the internals of the towers and is cooled by a counter-flow natural draft or by a forced draft created by fans. The water collects in the basin of the tower, where levels are monitored by level detectors. The level detectors provide high and low alarm signals if the level goes above or below the respective setpoints. The level detectors also provide a signal to an automatic city-water make up system whose function is to make up water lost by evaporation in the cooling process.

Forced draft cooling towers concentrate solids in the basin water and collect atmospheric dust. Hence, together with a filtration system, a feed-and-bleed purge is maintained while the towers are in operation in order to maintain proper water chemistry.

5.5.1.3 Heat Removal Considerations

The cooling towers are capable of rejecting 10 MW to the atmosphere under adverse (high humidity and temperature/weather) conditions.

5.5.1.4 Control and Safety Instrumentation

On/off controllers for the secondary coolant pumps are located in the reactor control room. There is no required safety instrumentation for the secondary coolant system because specifications associated with other heat removal systems are more limiting. However, as a backup, a scram signal is generated upon loss of secondary flow from either pump HM-1 or HM-1A. The locations of these flow scram instruments as well as that of other instruments are shown in Figure 5-4.

5.5.1.5 Radiation Monitors

Radioactivity could exist in the secondary coolant if there were a failure in either a primary or a reflector heat exchanger. Accordingly, secondary coolant is monitored for activity by redundant on-line gamma scintillation detectors. These would detect photons associated with nuclides such as N-16, F-18, and Na-24. They would not detect the presence of tritium. However, whenever the reactor is operating, there would be N-16/F-18 generated in both the heavy water reflector and the primary coolant and these would be detectable. Secondary coolant discharged to the sewer is continuously metered. Analyses for gamma, beta, and tritium-specific beta are performed daily during reactor operation and at least weekly during shutdown periods. This serves as a backup to the continuous monitoring available via the on-line detectors. In the event that activity were detected, cooling tower spray and bleed would both be secured. Also, the affected heat exchanger would be isolated.

5.5.1.6 Auxiliary Cooling System

The discussion in Section 5.5.1.1 of this report enumerates heat sources to the secondary coolant system.

5.5.1.7 Secondary Water Treatment System

The purpose of this system is to inhibit corrosion in the secondary system piping, to keep heat transfer surfaces clean, and to control the growth of bacteria in the system. A non-chemical treatment system is currently employed. A more conventional polymer-type and algaecide control system may also be used. Any method that is used will be evaluated based on visual inspection, thermal performance, corrosion coupon analysis, as well as routine chemical analysis.

5.5.1.8 Shutdown Cooling

The cooling towers are either aligned to spray or bypassed or isolated depending on the heat load. The shutdown flow path is for secondary coolant pump HM-1A to draw a suction from the basin and to discharge at a minimum flow rate through the secondary sides of heat exchangers HE-2 and HE-3 to the secondary return header. Alternatively, a small shutdown circulating pump may be used. A pressure switch is used exclusively during shutdown cooling. It shuts off the pump if the discharge pressure drops below its setpoint and thereby protects the pump against cavitation if flow is lost. The secondary sides of heat exchangers HE-1, HE-1A, HE-1B, and HE-D1 are isolated and vented to the main pump's suction.

The secondary flow through HE-3 is throttled upon reactor shutdown to avoid cooling the lead thermal shields too rapidly. Secondary flow to HE-3 may be completely secured after the reactor has been shut down for six hours provided that all non-shield heat loads have been secured.

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5-5 Flammability of Deuterium in Oxygen-Helium Mixtures, TID-208998, Explosives Research Center, Bureau of Mines, June 15, 1964.

5-6 File Memo (H_2/D_2 Concentration, June 1999)

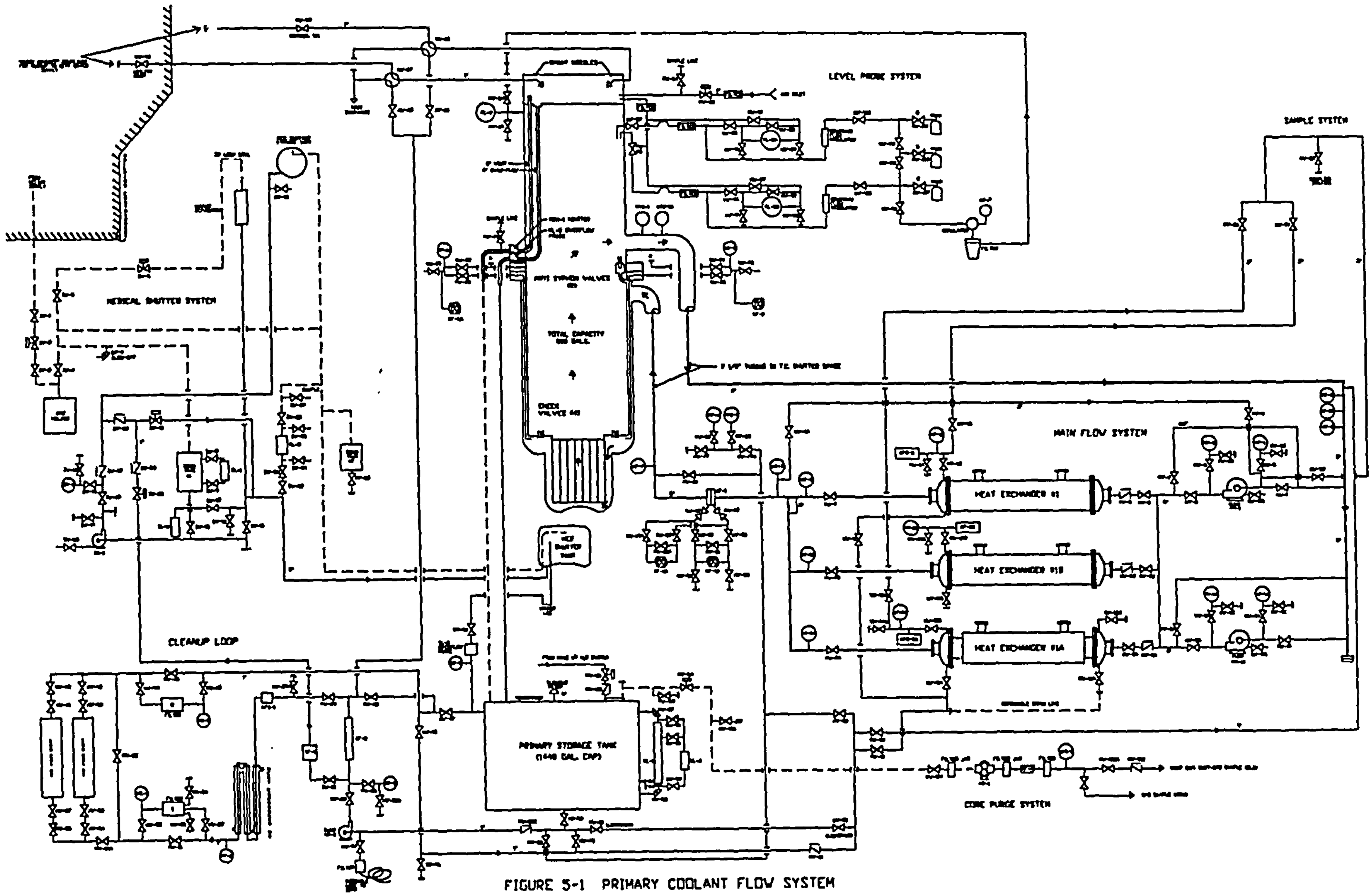

FIGURE 5-1 PRIMARY COOLANT FLOW SYSTEM

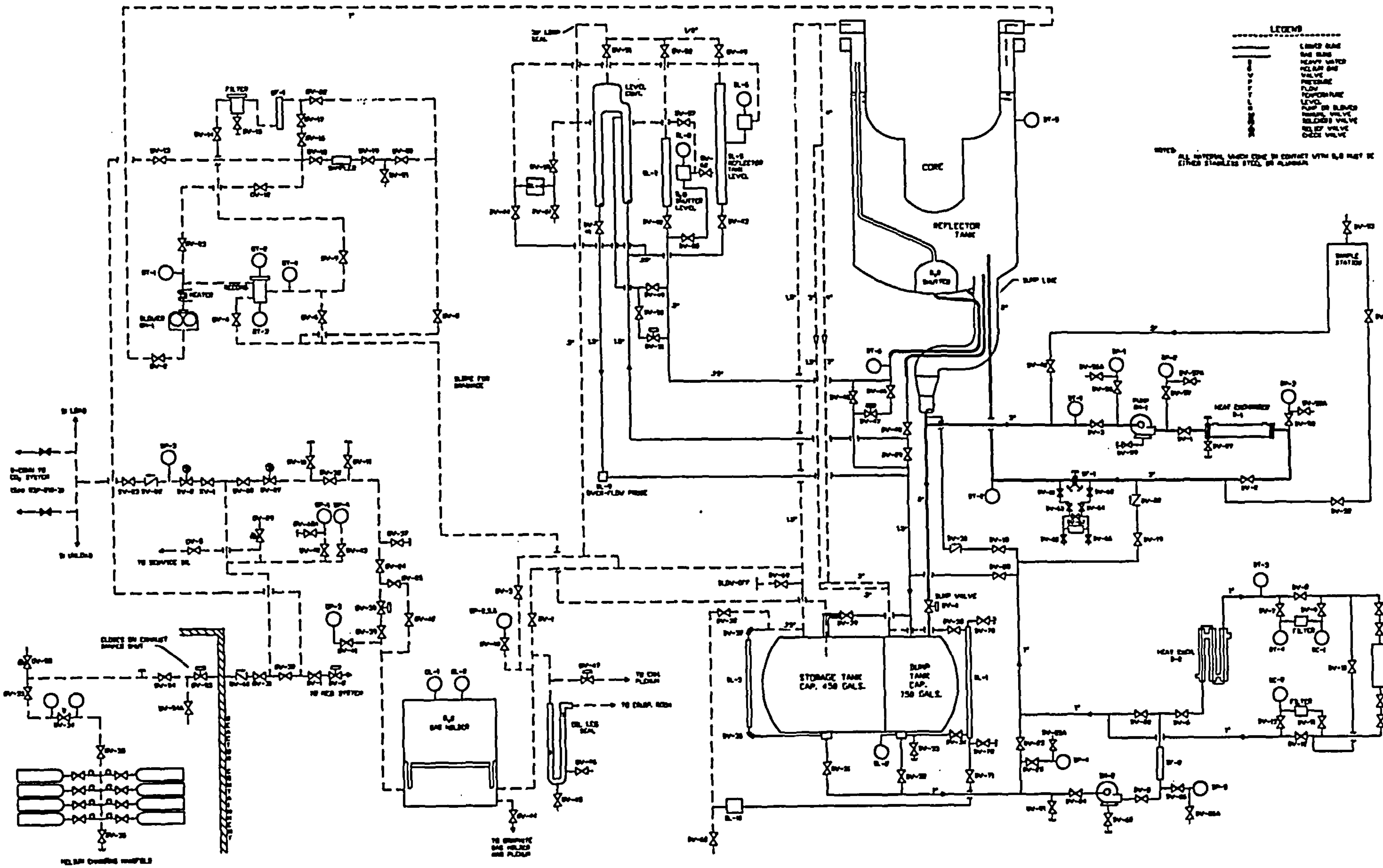

FIGURE 5-2 D₂O REFLECTOR FLOW SYSTEM

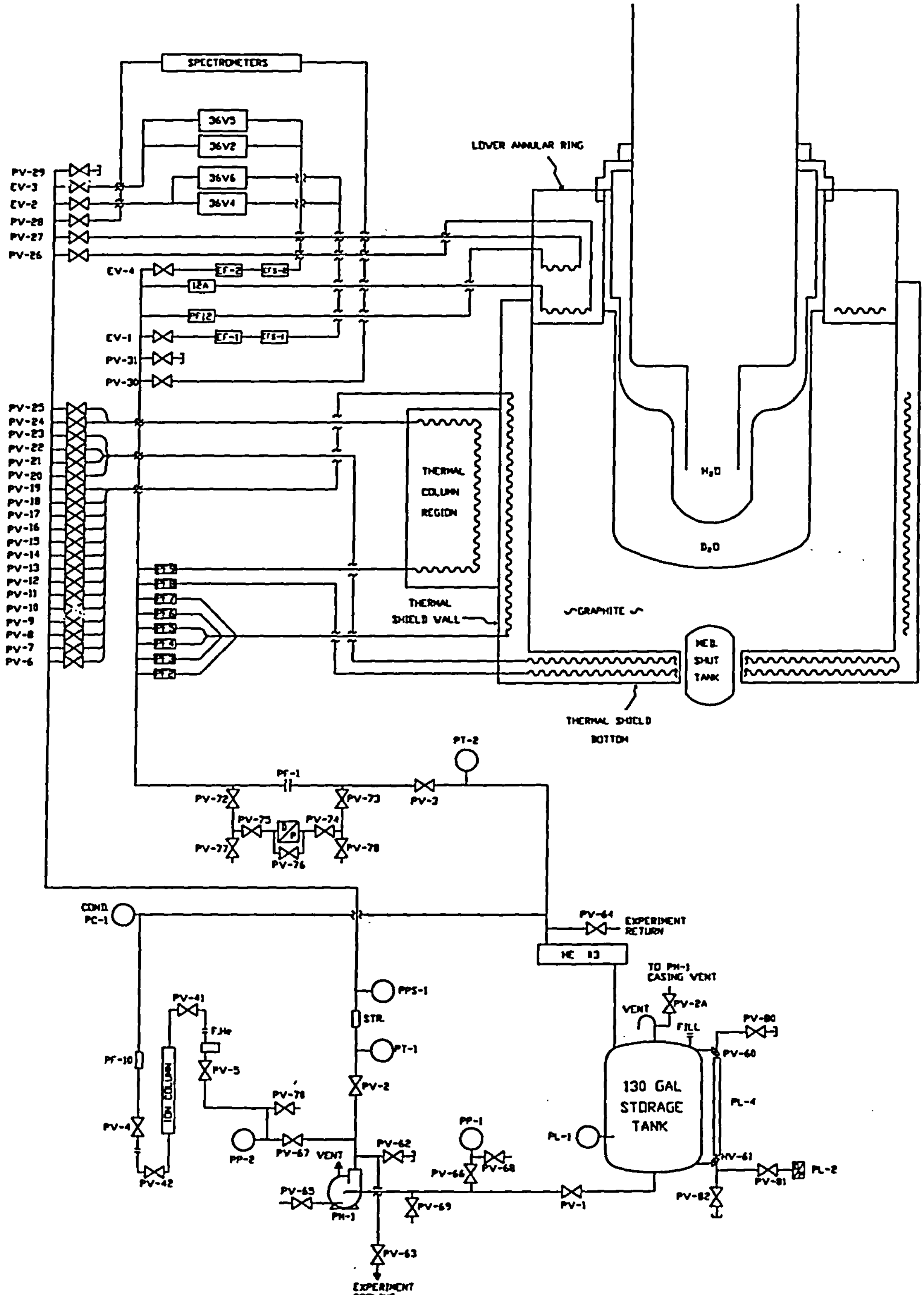

FIGURE 5-3 SHIELD COOLING SYSTEM FLOW DIAGRAM

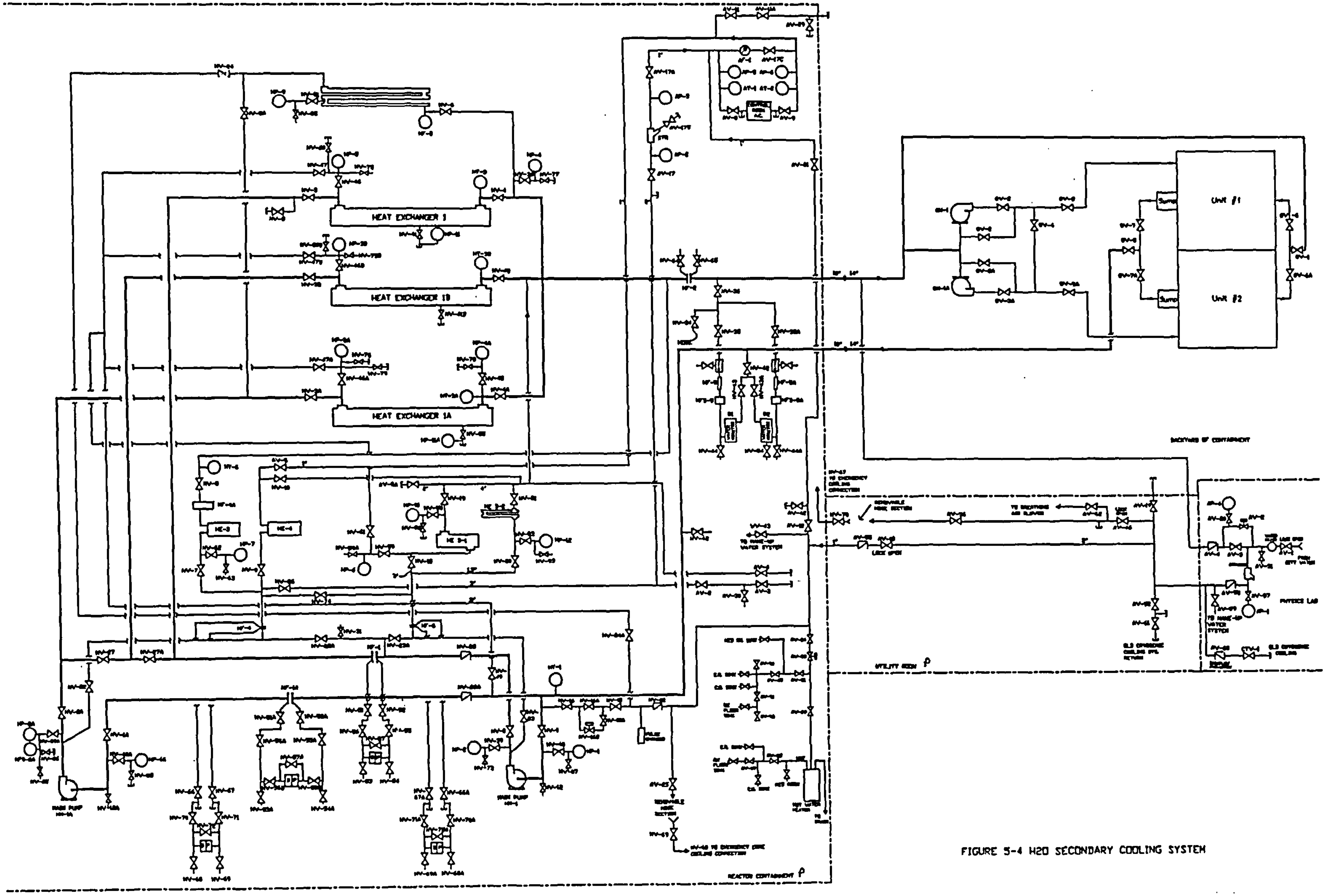

FIGURE 5-4 H2O SECONDARY COOLING SYSTEM

Chapter 6

Engineered Safety Features

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Chapter 6

Engineered Safety Features

6.1 Summary Description

The MIT Research Reactor's engineered safety features (ESFs) include natural convection and anti-siphon valves, an emergency core cooling system (ECCS), and the containment building. The latter is provided with protection against both vacuum and overpressure. All of the ESFs are passive. However, both the ECCS and the containment building's pressure relief system require manual initiation. The ESFs are one component of a defense-in-depth strategy of which the objective is to ensure reactor safety and to prevent the unauthorized release of radioactivity. The principal accident conditions against which the ESFs provide protection are overheating of the reactor core from decay heat in the event that forced convection flow is not available, loss of level in the core tank such that the core is not covered with coolant, and the inadvertent release of radioactive material.

a) Protection against overheating: Passive safety features, the natural circulation and anti-siphon valves, are used to promote the removal of decay heat from the core whenever forced convection flow is not available. The four natural convection valves are ball-type check valves that are located at the bottom of the core tank. These valves open upon loss of primary pump pressure (and hence flow) in the core tank inlet plenum and promote the re-establishment of natural convection flow by ensuring that coolant circulates through the core in the same direction as when forced convection exists.

   The two anti-siphon valves are also ball-type check valves. They too open upon loss of primary flow. These two valves are located at the inlet piping elevation and, when open, provide a second parallel path for natural convection cooling.

   The third ESF that is provided to protect against overheating during shutdown is the emergency core cooling system (ECCS). There is no credible accident that necessitates an ECCS. Nevertheless, one has been installed. The ECCS consists of two independent subsystems that provide water to two nozzles that are located above the core. The nozzles are positioned so as to provide a water spray to each fuel element. As discussed in Chapter Five of this report, the ECCS can be supplied either from the primary storage and reactor makeup tanks or from city water.

b) Protection against loss of level: The principal function of the anti-siphon valves is to protect against this scenario. The reactor core is contained in two concentric tanks. These are the primary and reflector tanks. Both tanks would have to rupture in order for the core to be exposed as the result of a loss of coolant accident. This is not considered credible. However, a siphon action in the primary tank could create the same situation with both tanks still being intact. Most of the primary coolant piping is physically below the core even though the inlet penetration to the core tank is above the core. Hence, a break on the inlet side of the primary system could create a siphon with coolant flow from the outlet plenum, down through the core, up through the inlet plenum (space between core-shroud and the primary tank wall), and down through the primary piping return line to the break point. The anti-siphon valves prevent this by breaking any siphon so that the coolant would remain at the inlet elevation, 1.8 m above the top of the core.

c) Protection against inadvertent release of radioactivity: The MITR is located within a gas-tight cylindrical steel containment building. Entry to the building is via personnel locks or a truck lock. Penetrations for ventilation are equipped with redundant quick-closing valves that are instrumented so that, upon detection of any abnormal airborne radioactivity, the valves close before that activity can be released. The containment building is designed to withstand the overpressure that might be generated during credible accidents. Also, it is protected against greater overpressure by a pressure relief system and underpressure by two sets of vacuum breakers. The pressure relief system is manually initiated. This is done from outside the reactor containment building. Once the pressure relief system is on line, containment building air is discharged to atmosphere via filters that remove particulates and iodine. The vacuum breakers are held closed by springs. They open automatically if atmospheric pressure exceeds that of the building by a preset amount.

All of the ESFs are subject to surveillance tests to ensure proper operation.

6.2 Natural Convection Valves

There are four natural convection valves located at the bottom of the core tank as shown in Figure 6-1. The purpose of these valves is to promote removal of decay heat by natural convection should forced convection not be possible.

Figure 6-2 shows the design and operation of a natural convection valve. The body of the valve is made of 101.6 mm (4") diameter, Al-6061 T6 round pipe. The solid ball, which has a diameter of 73.0 mm (2.87"), is made of the same material. The surfaces of the valves have been hardened either by Sanford coating or by work-hardening. If the pressure below the valve is sufficient, the ball is forced upwards thereby closing off flow. When the pressure drops, the ball falls and the valve opens. Natural convection flow is then established within the core tank because of the buoyancy force of the heated coolant in the core region. The hot coolant exiting the core rises within the core tank, mixes with cold coolant in the outlet plenum, reverses direction and flows through the natural convection and/or anti-siphon valves, and then goes back through the core region thereby completing the natural circulation loop.

Analysis has shown that either three of the four natural convection valves or the anti-siphon valves alone (Section 6.3 below) are enough to remove the decay heat from 6 MW steady-state operation [6-1]. The redundancy of the valves assures that there is more than enough cooling for the core during a loss of flow accident.

6.3 Anti-Siphon Valves

Another two check valves are installed in the core tank at the elevation of the primary inlet pipe. Like the natural convection valves, they are designed to open upon loss of primary pump pressure. The purpose of the anti-siphon valves is to prevent the core tank from complete drainage should the primary inlet piping rupture. However, these valves also promote natural convection cooling by providing a second flow path whereby cooler coolant from the outlet plenum can flow under the core and up through it.

The axial location of the anti-siphon valves is shown in Figure 6-1. Figure 6-3 shows the design and operation of an anti-siphon valve. The chambers of the anti-siphon valves are made of 76.2 mm (3") diameter, Al-6061 T6 pipe. There are eighty-four 12.7 mm (1/2") holes drilled in each chamber. These holes allow air to enter from the top of the valve to the inlet annulus and thereby break any siphon. The solid balls are made of Al-6061 T6 and are Sanford hard-coated. The diameter of the solid ball is 58.7 mm (2.31"). Screens are installed on the tops of both valves to prevent foreign objects from entering the valves and causing inadvertent blockage.

The anti-siphon valves are normally closed during reactor operation because of primary pump pressure. The solid ball starts to drop when the primary flow decreases to a point where the inlet pressure is not enough to hold the ball. If the inlet pipe were to rupture, air will enter the inlet annulus through the anti-siphon valves when the coolant level drops and thus break the siphon.

Operation with only one anti-siphon is sufficient to break the siphon loop. The second valve was installed for redundancy.

6.4 Emergency Core Cooling System

A scenario that requires an ECCS for the MITR is not considered credible because the core is contained within two concentric tanks (the primary and reflector tanks) and siphoning is prevented by passive safety features. Specifically, a leak from the core tank itself will result in mixing of the light-water coolant and heavy-water reflector and not in a major loss of coolant from the core tank. Also, the helium vent lines at the top of the reflector tank and above the reflector level sight glass are designed to prevent the possibility of siphon action and hence to limit the loss of water to the level of the primary coolant inlet and outlet pipes. (See Section 13.2.3.2 of this report.) Similarly, a rupture of the primary inlet piping would not cause a major loss of coolant because the anti-siphon valves would stop the siphon effect and stop the coolant loss at 1.8 m (6.0 ft) above the top of the fuel plates. For both cases, natural convection cooling is sufficient to remove the decay heat from the core region with a pool of water remaining above the core. A complete loss of coolant accident requires one of the following incredible scenarios:

a) A simultaneous rupture of the core tank and the reflector tank.

b) A rupture of the inlet pipe below the core level, together with a failure of both anti-siphon valves to open.

c) A core tank puncture caused by an explosion at one of the beam port reentrant thimbles that extend inward from the reflector tank. (See Section 13.2.3.3 of this report for analysis of this scenario.)

Figure 5-1, which is a schematic of the primary coolant system, includes a diagram of the ECCS. A detailed description of the ECCS is given in Section 5.2.5 of this report. To summarize, there are two independent subsystems, each with its own spray nozzle. Water for the system is available from either of two sources: city water or the primary storage/makeup water tanks. The possible lineups are:

a) City Water: Flow is initiated by use of either of two quick-connect hoses, each with special couplings and a pair of valves. The source of the flow is city water pressure. No pump is needed. There are two possible initiation points for this flow. One is in the reactor control room. The other is at a location outside the containment building. Hence, it is possible to initiate ECCS without entry to the reactor building. (Note: The use of the quick-connect hoses minimizes the potential for the inadvertent introduction of chlorinated water to the core tank during normal operation. It also protects against the mixing of primary coolant with potable water. However, that scenario is already precluded because all water supplied to the containment building first passes through a backflow preventer.)

b) Storage Tanks: Flow is initiated using the primary coolant system's auxiliary pump, MM-2. Water would be drawn from the primary storage tank which would in turn be replenished from the makeup water tank.

The ECCS system is redundant in that flow through either of the two nozzles is sufficient to protect the core. The spray nozzles are located above the core and are positioned such that each fuel element will receive at least 20% of the average flow. A series of full-scale out-of-core mockup tests were performed for various in-core experiment configurations. Results showed that the 20% average flow criterion was met for the current nozzle design [6-2].

The emergency spray system is normally lined up to the city water system, which is the preferred supply. It must be revalved in order to use the other source of supply.

Calculations were performed to analyze the capacity of the ECCS spray system for the MITR. The system design is based on the initiating event being an experiment malfunction in one of the beam ports, which causes a break of 63.5 mm (2.5") ID in the core tank. This analysis includes the following:

a) The estimated time to drain the core tank was calculated using the Bernoulli equation.

b) The decay heat at the time of drainage was calculated using DKPOWR which is a LANL code based on ANSI/ANS 15.1-1979, "American National Standard for Decay Heat Power in Light Water Reactors" [6-3]. It was assumed that the reactor had been operating at 6 MW long enough to reach saturated fission product concentrations. It is also assumed that 87.5% of the decay power is deposited in the fuel [6-4]. The remaining 12.5% is deposited in the reflector and biological shield.

c) A check was made to determine if the decay of activated aluminum would contribute to the heat load.

d) An evaluation of the flooding limit was made to ensure that the cooling was adequate.

A 20% flow distribution factor was used to take into account the minimum flow into each element. A safety factor of 2.0 was adopted to ensure an adequate safety margin. Thus,

$$\dot{m}_{design} = \frac{1}{20\%} \times 2.0 \times \dot{m}_{calculated} \qquad (6\text{-}1)$$

where $\dot{m}_{calculated}$ is the calculated flow rate to remove decay heat by evaporation from the average channel, and $\dot{m}_{design}$ is the design flow rate.

It was determined that the time to drain the core tank was 411 seconds (starting from 4" below overflow). The power produced from decay heat five minutes after scram is 148 kW. The decay power from aluminum activation by the fast neutron flux was found to be negligible (0.2 W). The ECCS design flow rate was thus 9.5 gpm [6-5].

To assure that this design flow rate was capable of cooling the fuel, the flooding limit was also analyzed. Flooding refers to the stalling of a liquid downflow by a sufficient rate of gas upflow [6-6]. The spray will not be able to enter the hot channel if the flooding limit is exceeded (i.e., predicted hot channel vapor flow rate greater than that of the flooding condition). The calculated result showed that the vapor velocity in the hot channel was lower than that of the flooding limit by a factor of four.

The effect of a non-uniform flow distribution within a fuel element was also evaluated. This calculation assumed that one of the fuel plates was not wetted by water and that the only effective heat removal path was conduction to the side plate, which is cooled by the flow in the adjacent channels. The calculation assumed that, for the worst case, the flow rate in the adjacent channels was 20% of the average flow. The calculated maximum fuel centerline temperature was about 207° C, which is far below the fuel clad softening point of 450° C.

6.5 Containment

6.5.1 Description

The containment building is a domed cylindrical structure with an outside diameter of 22.5 m (74 feet) and a height of 14.9 m (49 feet) between grade elevation and the top of the dome. The containment shell is constructed of 9.5 mm (3/8 in.) thick steel plate (15.8 mm (5/8 inch) in the dome) welded to form an air-tight structure with the external temperature ranging from -20° F to 110° F. The containment is designed to withstand an internal pressure of 1.38×10^4 Pa (2 psig) greater than atmospheric and 690 Pa (0.1 psig) less than atmospheric. The building is maintained at a slight vacuum.

Contained within the circumference of the steel shell is a cylindrical concrete wall that is 2.0 feet thick and 31.5 feet high. This wall provides additional radiation shielding to reduce the dose rate at the outer surface of the reactor building. The wall also supports a polar crane which has both a 20-ton and a 3-ton hook suspended from a movable bridge that spans the interior of the containment building.

There are two floors, referred to as the main and basement levels, within the building. The reactor is situated in the center of the main floor so as to provide a 25-foot wide annular ring of floor space around it for the placement of experimental apparatus. The design loading of the main reactor floor, extending a distance six feet from the reactor face, is 3000 lbs/ft^2. The remaining floor area is rated for a loading of 2,000 lbs/ft^2. A stairway leads to a catwalk that runs along the inner wall of the building. A removable bridge provides access from the catwalk to the reactor top. The catwalk is rated for a maximum load of 100 lbs/ft^2 with a 1500 lb total load capacity. An electric lift is provided to transport sample containers from the reactor floor to the outer catwalk.

The reinforced-concrete basement contains the control room, equipment room, primary chemistry room, secondary chemistry area, the basement medical therapy room and control area, spent fuel storage room, and a general set-up area. The basement floor loading is restricted to 2000 lbs/ft^2 with a total permissible load of 100 tons.

An unusual type of foundation support was required because the water table at the facility site is only slightly below ground level. The entire building is supported by a 3.5-foot thick foundation pad which floats without additional attachment to a sub-foundation. Gas-tight seals are made at holes for an elevator, an operating-table hydraulic lift in the basement medical therapy room, and the spent fuel storage tank. Below-grade inside surfaces are water proofed with Ironite. Concrete cantilever beams extend from the outer basement wall to support the main personnel airlock.

Openings through the three-foot thick reactor floor to the basement include the stairwell, the service elevator, a five-foot diameter plugged hole over the fuel storage room, an eight-foot diameter plugged hole over the equipment room, and another eight-foot diameter plugged hole over the basement setup area. The eight-foot diameter openings permit maintenance operations that require use of the crane to be performed in the basement area. The five-foot hole over the fuel storage room allows containers such as the fuel element transfer cask or a spent fuel shipping cask to be lowered to the storage tank for addition or removal of spent fuel elements. Numerous small pipes and openings also penetrate the reactor room floor around the face of the reactor and near the outside wall. These allow instrument cables, fluid lines, and other service connections for experiments to run under the reactor floor from the face of the reactor to the outer wall.

6.5.2 Design Specifications

The steel chosen for the shell is ASTM 283 Grade C. The shell as constructed meets the American Petroleum Institute specification titled, "Welded Oil Storage Tanks" which is API Standard 12C (1958). It also meets specification API-620 (1958), titled, "Recommended Rules for Design and Construction of Large Low-Pressure Storage Tanks." Finally, the building has been tested successfully at 1.38×10^4 Pa (2 psig) internal pressure. This test, as described in

Section 6.6, was first performed by the builder and subsequently by the MITR staff. The maximum permissible leakage rate is 1% of the building volume per day per psi of building overpressure.

6.5.3 Building Penetrations

Building penetrations are listed in Table 6-1. The intake and the exhaust air ducts remain open during normal operation of the ventilation system. These air ducts can be sealed rapidly at any time by means of fast-closing hydraulically-actuated butterfly dampers. A gas-tight seal is maintained at each of the other penetrations with certain exceptions. The exceptions involve the compressed air system, air lines for the personnel locks, and certain small instrument lines. The justifications for the deviations are as follows:

a) Compressed Air: The compressed air system is a closed system designed for pressures greater than 1.38×10^4 Pa (2.0 psig). If a line should break either inside or outside the containment, this single failure would not result in a breach of the containment because there are check valves in all lines that penetrate the containment building shell.

b) Personnel Locks: Pressurized air is used to inflate the seals in the personnel lock doors and in the truck lock. The use of a "fail safe" automatic closing valve in the air lines that penetrate the building containment might lead to the more serious problem of loss of air pressure to recharge the reserve tanks on the door seals. Therefore, the isolation valves are normally open and manually operated.

c) Instrumentation: Several small instrument lines do not have isolation valves but are designed so that containment leakage resulting from a complete rupture of a single line would not exceed 3% of the allowable building leakage of 1% of volume per day. These instrument lines are widely separated so that coincidental failure is considered incredible. The acceptance criteria for the building pressure test is 95% of the allowable leakage so that a single instrument line will not result in leakage in excess of allowable. An example of instrument lines that are within this category are those for the measurement of building differential pressure. (Note: The instrument air lines for the containment building heating and air conditioning system meet the above criteria for instrument lines and, in addition, are provided with manual isolation valves outside of and close to the containment penetration.)

Table 6-1

Reactor Building Penetrations

Above Grade No. of Penetrations	Description
1	4" pipe at chopper window
1	24" aluminum chopper window
1	4" aluminum chopper window
1	2" air conditioning effluent pipe to waste shed
2	2-1/2" air conditioning coolant pipe from utility room
1	30" inlet air duct from utility room
1	1" cold water supply (includes ECCS) from utility room
2	2" pressure test lines to utility room
1	1" pressure test line to utility room
1	Personnel lock 5' x 8' door
1	Truck lock 10' x 14' door
2	10" vacuum breaker lines above personnel lock
1	1" makeup water return line
1	2" pipe (capped spare)
1	3" pipe (capped spare)
1	4" pipe with coaxial and signal cables
1	5" pipe with smaller lines for ECCS, air conditioning controls, utility room ΔP gauges, makeup water supply, and spares from utility room
1	1/2" pipe ΔP line to waste shed
3	1" Connax seals to back stairwell
2	2" pipes (capped spares) to utility room

Table 6-1 (Cont.)

Below Grade No. of Penetrations	Description
3	4" capped pipe sleeves in basement at 120° intervals
1	1-1/4" water effluent pipe from sump to back stairwell
2	14" pipe lines for secondary coolant system
1	30" air effluent duct to base of stack
1	3" electric pipe to base of stack
4	1" multiconductor electrical feed through fittings to base of stack
2	Electrical coaxial feed through fittings to base of stack
1	Basement personnel lock 3' x 7' door to back stairwell
1	1-1/2" pipe sleeves for pneumatic tube to utility room
3	1-1/2" spare pipe sleeve (capped) to utility room
2	4" conduits for control and intercom wires to the utility room
1	4" pipe with smaller lines for gas and ΔP from utility room
2	3" pipe for building pressure relief system to base of stack
1	6" air conditioning coolant and control lines to back stairwell
3	Vertical penetrations through foundation pad: (1) fuel storage pool; (2) elevator hydraulic lift; (3) medical therapy room hydraulic lift – gas tight seals.

Electric Service No. of Penetrations	Description
4	3" conduit power wiring from utility room
1	1" conduit emergency a.c. power from utility room
1	2" conduit emergency d.c. lighting from utility room
1	2-1/2" conduit (experimental receptacles, panel 1, emergency power monitor) from utility room
2	1-1/4" conduit for telephone from utility room
9	3/4" conduit for control wiring from utility room

6.5.3.1 Aluminum Window

A 24-inch square, thin aluminum window, opposite the 12-inch horizontal port, can be used to project a neutron beam through the containment shell, thereby allowing the beam's flight path to extend as far as the facility's perimeter wall. The window is constructed of a 1/8-inch sheet of aluminum that is mounted in a 24-inch square flange that penetrates the shielding wall and the steel shell. A collar is seal-welded to both the window frame and the containment shell in order to maintain the gas-tight integrity of the containment. A 5/8-inch thick steel door, hinged to a flange that is bolted to the outside of the building, can be swung closed if the aluminum window ruptures. When this door is bolted securely to a gas-tight surface, a tight building seal is ensured. Adjacent to the 24-inch window is a 4-inch diameter, 1/16-inch thick aluminum window, which can be used to transmit a neutron beam from the 12-inch port.

6.5.3.2 Main Personnel Airlock

Access to the main floor of the reactor building is through the main personnel airlock which enters at the west side of the containment building. The airlock consists of a gastight steel passageway which is sealed at each end by steel doors that provide a five-foot by eight-foot clear opening. Both doors swing inward toward the reactor, providing ten feet of clear length within the sixteen-foot long airlock. When closed, the doors form a gastight seal by means of inflatable rubber gaskets that bear on the door jambs. The doors are normally operated by hydraulically-actuated mechanisms which are controlled by push-button stations located on either side of each door. An electrical interlock prevents one door from being operated electrically while the other is unsealed. A redundant mechanical interlock provides additional protection against having both doors open simultaneously. In the event of an electrical or mechanical malfunction in the door-operating mechanisms, the doors can be manually operated, one at a time, by turning a hand wheel at either side of each door. This action either deflates or inflates the gasket through a linkage to the actuating mechanism. The hand wheels are also interlocked mechanically so as to allow only one door to be unsealed at a time.

The gaskets are normally inflated by air from the compressed air system's main compressor. If that unit fails, air is automatically supplied from a backup compressor through a check valve. A bottle of compressed air is permanently installed in this airlock's control air system as an additional backup for use in emergencies. An alarm annunciates in the control room whenever the compressed air supply to the inflatable door gaskets falls below 20 psig. In addition to this alarm, a second alarm and an automatic reactor scram will occur if both airlock door gaskets are deflated simultaneously. The personnel airlock is also equipped with a check valve to relieve excess pressure within the lock to the reactor building.

6.5.3.3 Truck Airlock

A truck airlock that is located at the northeast side of the containment shell enables large equipment to be transported into the reactor building. This airlock, which is approximately 10 feet wide by 12.5 feet high by 25 feet long, employs vertically sliding doors to close off each end. A check valve is installed in the inner door to relieve overpressure within the truck lock to the reactor building. In order to allow for thermal expansion, the lock was constructed in two sections with an inflatable rubber gasket sealing the expansion joint and thereby maintaining the truck lock's air-tight integrity. The air for the gasket is supplied from the same air system used in the personnel airlock. Electric motors located above the lock openings move each door by means of a chain and sprocket drive. The doors are counter-balanced to prevent them from dropping in the event of a power failure. An electrical interlock is provided to prevent both doors from being opened simultaneously. When the door is closed, hydraulically-actuated rams force the door against rubber gaskets to create a gastight seal.

6.5.3.4 Basement Airlock

A small personnel airlock is located below grade in the basement to provide an exit from the control room to the backyard. The gasketed steel doors swing in toward the reactor thereby providing a three-foot by seven-foot clear opening. This airlock is manually operated with a mechanical interlock to prevent both doors from being open at the same time. The doors are sealed using both a rubber gasket and the same compressed air supply that is used for the main personnel lock. A small vacuum blower provides suction on the gaskets when they are deflated. This makes operation of the doors easier. As is the case with the main personnel airlock, an alarm and an automatic reactor scram will occur if both door gaskets are deflated simultaneously. A check valve will relieve pressure in the lock to the reactor building.

6.5.4 Pressure Protection

6.5.4.1 Protection Against Vacuum

Two sets of vacuum breakers are installed to protect the integrity of the containment in the event of an excessive underpressure within the building. Each consists of an interior and an exterior breaker in series. The interior breakers are set to open when the internal pressure is between 100 and 250 Pa below atmospheric (-0.015 and -0.036 psig) while the exterior breakers are set to open between 250 and 430 Pa below atmospheric (-0.036 and -0.062 psig).

6.5.4.2 Protection Against Excessive Pressure

If the building pressure should approach its rating of 1.38×10^4 Pa (2.0 psig), a safe, effective relief can be achieved by use of the pressure relief system. This system, which is shown in Figure 6-4, filters the exhaust air and discharges it to the base of the ventilation exhaust stack above the stack base damper. All parts of the relief system, including this damper, are manually operated. The system exhaust line contains two high-efficiency absolute particulate air filters that are 99.9% efficient for particle sizes of 0.3 microns, and an activated charcoal filter that is 99% efficient for removal of elemental iodine. The charcoal filter is positioned between the two absolute filters. The first absolute filter protects the charcoal filter from any particles that might be drawn into the relief system while the second absolute filter removes any flakes of charcoal that may have broken off. The charcoal filter absorbs any iodine that may be present in the exhaust air. The rated system flow is 800 cfm, but if the system is used, actual flow would be determined by the difference between the internal building and atmospheric pressures and would be considerably below the rated capacity. A pressure relief blower is used to draw air through the system in order to both clean and activate the charcoal.

6.5.5 Building Isolation

The ducts for the containment building ventilation system are automatically sealed upon detection of an abnormal radiation level in the exhaust air. A description of this feature is given in Section 9.1.2.6 of this report.

6.5.6 Shadow Shielding

In addition to the two-foot thick shield around the inside perimeter of the containment shell, the personnel airlock and the truck lock are shadow-shielded with concrete. An eighteen-inch thick concrete wall along the east perimeter of the site, opposite the truck lock and the neutron window, provides additional shielding.

6.6 Test and Surveillance

Natural convection valves and anti-siphon valves are passive designs which operate as a function of the primary flow condition. These valves are checked by the operator as follows to ensure proper operation:

a) Checked closed when the primary pumps are turned on for reactor startup above 100 kW. This check is done to ensure that a valve does not remain open and allow excess bypass flow.

b) Checked open after shutdown once primary pumps are turned off. This check is done to ensure that a valve does not remain shut and preclude adequate natural circulation cooling.

The check may be done visually or audibly through use of an instrument, such as a stethoscope.

The operability and the discharge rate of the emergency core cooling system are checked through surveillance testing. The acceptance criteria of the test are:

a) Uniform spray from the nozzles,

b) Flow rate of city water through the test nozzles is greater than the design flow rate (9.5 gpm), and

c) Flow rate from the auxiliary pump through the test nozzles is greater than the design flow rate (9.5 gpm).

Surveillance testing of the containment includes a building leakage test, an efficiency test of the pressure relief system, a test of the vacuum relief breakers, and an inspection of the ventilation isolation valves.

An integral air leakage rate test of the reactor building containment is performed once every two years. Leak tests of individual penetrations are done between integral tests whenever new penetrations are made or repairs of existing penetrations are necessary. The sum of the results of the last integral building test and any increase in the penetration leakage since the most recent integral test must be less than 1% of the building volume per day per 6895 Pa (1 psi) of building overpressure.

Proper functioning of the independent vacuum breakers is verified annually. The isolation dampers are inspected annually. A test of the charcoal filters in the pressure relief system is performed annually to determine their efficiency for the removal of elemental iodine. The filters are replaced if the efficiency is 95.0% or less.

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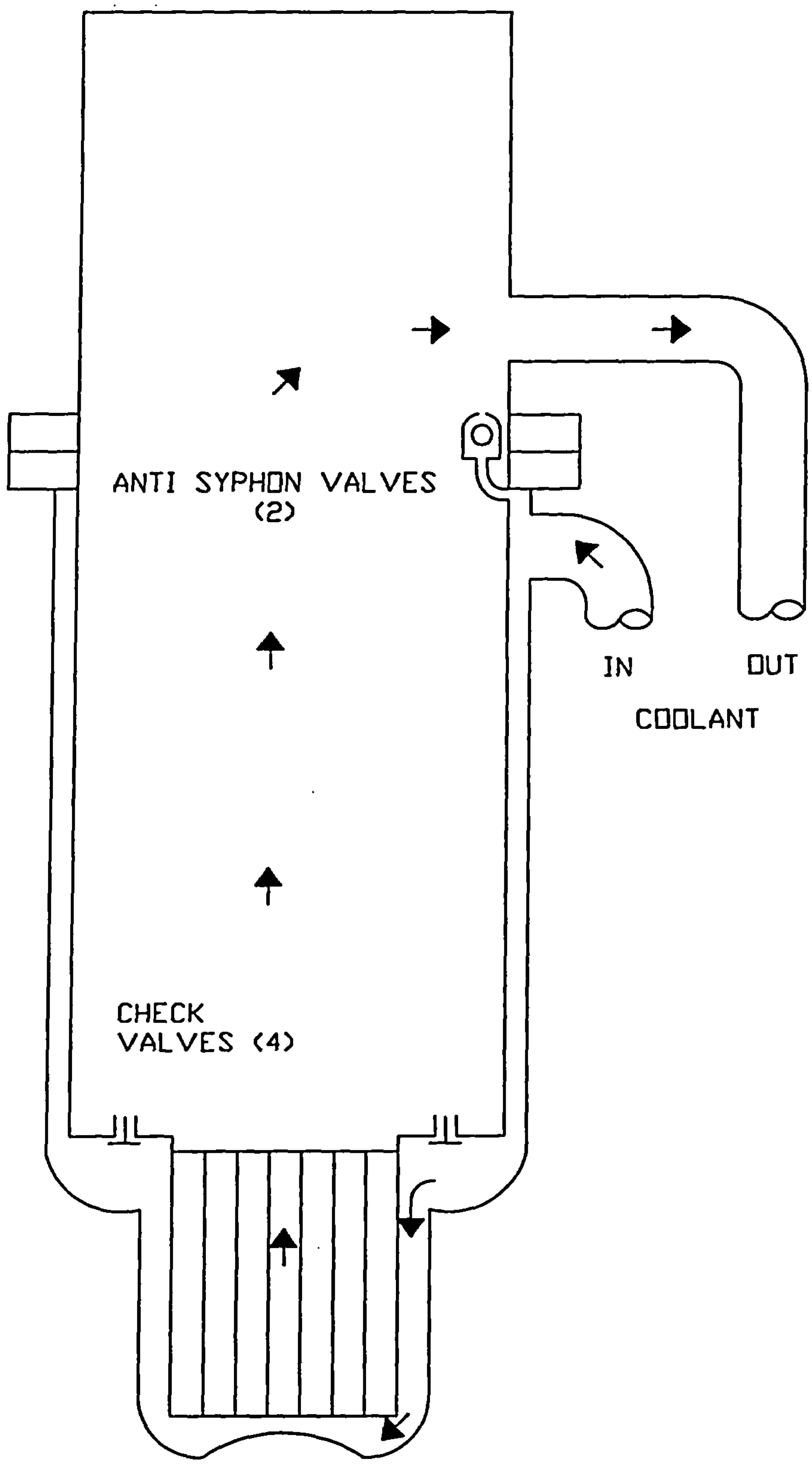

LOCATION OF CONVECTION AND ANTI-SIPHON VALVES

FIGURE 6-1

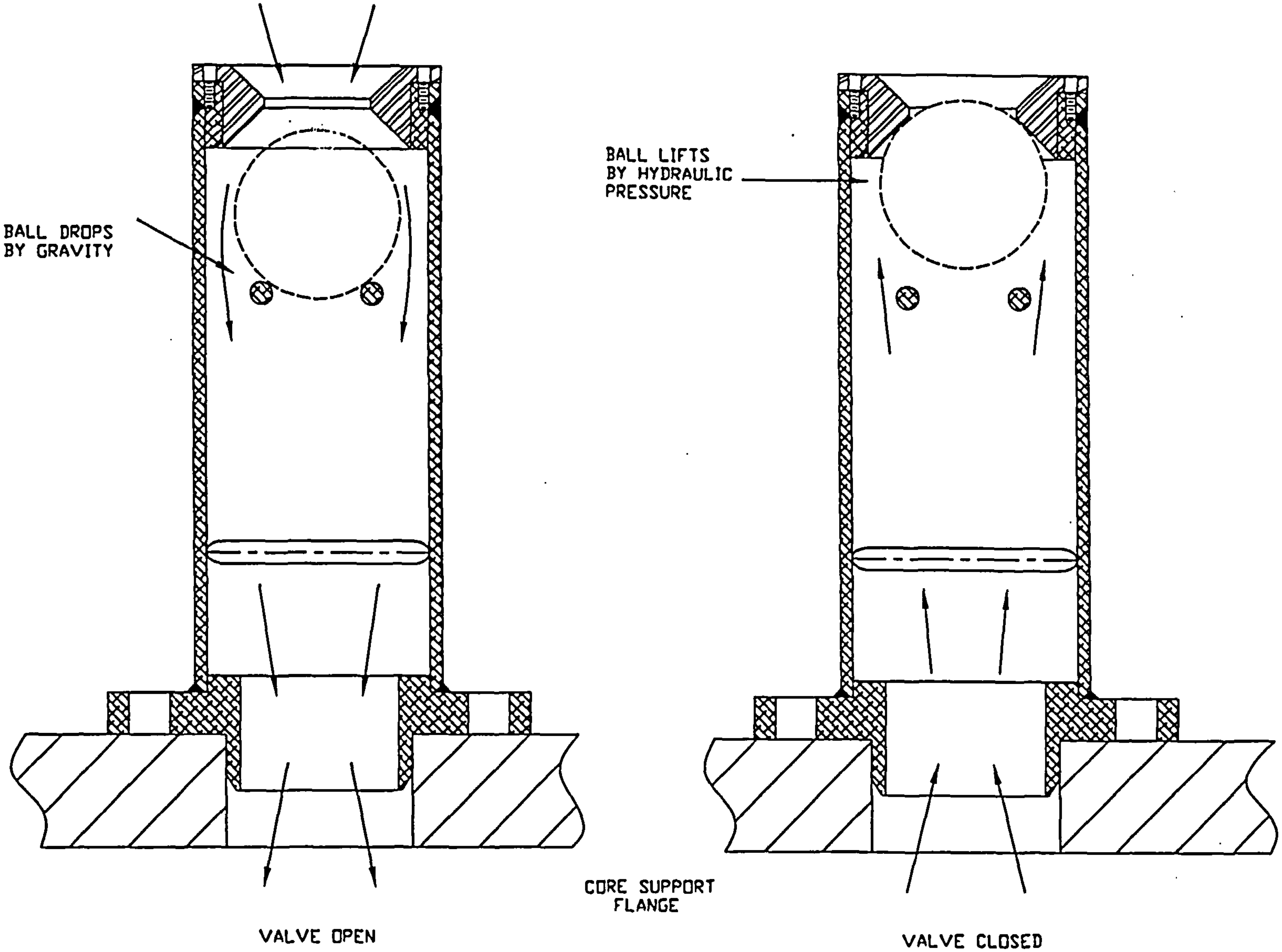

FIGURE 6-2
NATURAL CONVECTION VALVES

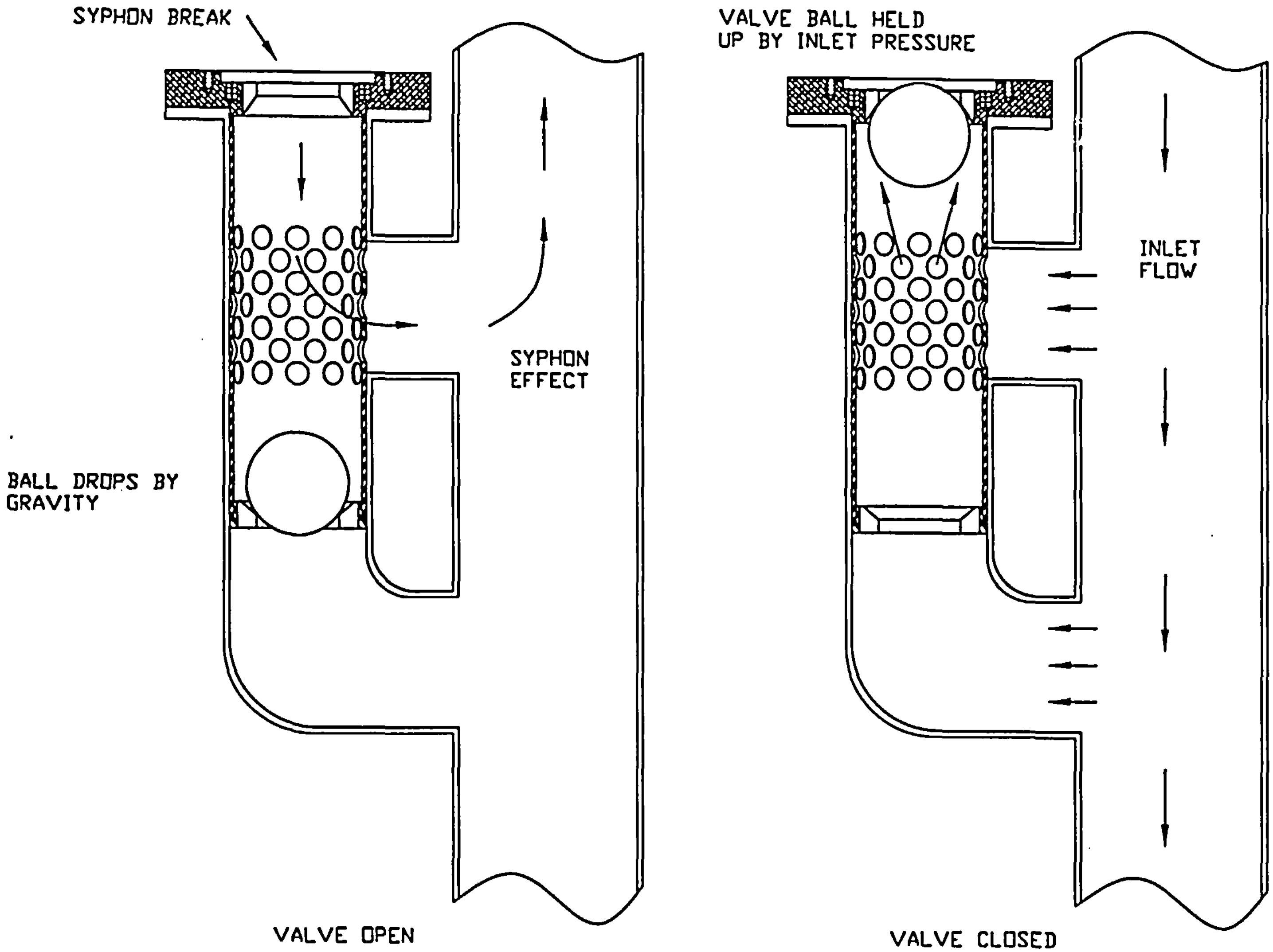

FIGURE 6-3
ANTI-SYPHON VALVES

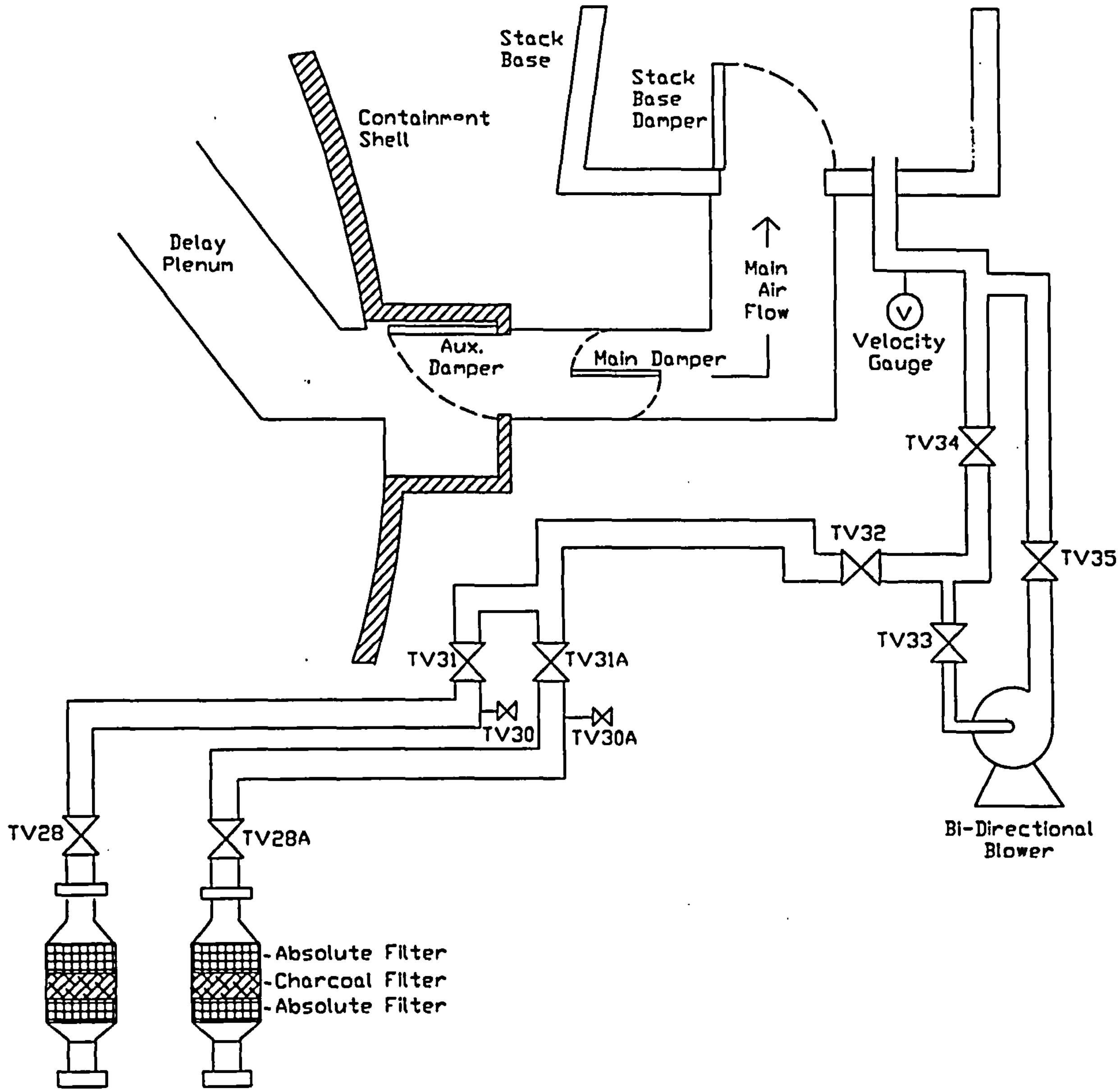

FIGURE 6-4
PRESSURE RELIEF SYSTEM

Chapter 7

Instrumentation and Control Systems

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Chapter 7

Instrumentation and Control Systems

7.1 Summary Description

The MITR instrumentation and control system consists of five major subsystems. These are the reactor control system (RCS), the reactor protection system (RPS), the engineered safeguards actuation system, the console display instrumentation, and the radiation monitoring system (RMS). Each of these subsystems may be further subdivided. For example, the RCS has both a manual and an automatic capability. The RPS has both a nuclear and a non-nuclear component, and the RMS has both an area (internal to the containment building) and an effluent monitoring capability.

The MITR was originally built in 1958 and all instruments were therefore analog. Many of those were replaced during the 1974/1975 modification, again with analog units. Subsequent upgrades were made in preparation for the MITR-III, which is the subject of this report. These instruments have been mostly digital or a hybrid of digital and analog technology because the digital design offers greater flexibility. The nuclear safety system, which is the part of the RPS described in Section 7.4.1, and the radiation monitoring system's effluent monitors remain entirely analog. The area radiation monitors are digital hybrids as are some of the instruments used for reactor control and console display. The term "digital hybrid" as used here implies an instrument that obtains an analog signal, transmits that signal to an analog-to-digital converter, and then makes the digital output available for display or other desired use. Setpoints and alarms may be altered by authorized personnel on both the analog and the digital-hybrid instruments. However, the programming of the digital-hybrid instruments cannot be changed by the MITR staff. It is embedded by the manufacturer in the instrument and is not alterable.

7.1.1 Reactor Control System

The reactor control system or RCS permits the MITR to be operated in either a manual or an automatic mode. In manual mode, the console operator can manipulate the six shim blades and the fine-control regulating rod so as to start up the reactor, change the power level, or shut the reactor down. In automatic mode, the regulating rod's position is adjusted by an automatic controller so that a particular power setpoint is maintained. The MITR is not equipped with a digital console although, as described in Chapter 10 of this report, the RCS can be modified to permit the conduct of experiments that test new methods for the digital control of nuclear reactors.

The instruments used by the reactor control system are distinct from those used by the reactor protection system. The RCS is equipped with a number of design features and/or interlocks to ensure safe operation. The most important ones are that only one shim blade can be withdrawn at a time and that receipt of any scram signal causes all control device drives to be driven in. (Note: The six shim blade absorbers will, of course, already have been dropped into the core by the reactor protection system.) The MITR does not have a pulse capability.

7.1.2 Reactor Protection System

The reactor protection system or RPS consists of a nuclear and a non-nuclear component. The former monitors both the reactor period (three channels) and the reactor power level (three channels). The latter monitors process parameters which, together with the reactor power, define the limiting safety system settings . These parameters are the primary coolant flow, the primary coolant temperature, and the core tank level. In addition, all parameters that are required by the technical specifications to cause a reactor scram have an input to the reactor protection system. This is achieved through the "withdraw permit" circuit which is a series of relays with each relay corresponding to a scram condition. If any relay is open, electric power is interrupted to the shim blade electromagnets, which scrams the reactor.

The actuating logic for the RPS is that any one sensor indication that exceeds its setpoint will cause a scram. With one exception, there is no use of a two-out-of-three logic even though the most important parameters are monitored by three independent sensors. The exception is the off-scale level trip for the period channels which is a two-out-of-three logic. (See Section 7.4.1.1 of this report.) Once the actuating logic has caused a trip, the withdraw permit circuit will open and power is interrupted to all six shim blade electromagnets, and the blades are released to drop into the reactor and shut it down.

7.1.3 Engineered Safeguards Actuation

Most of the MITR's engineered safeguards features are either passive or manually initiated. There are two exceptions. The first is the containment building isolation system, which is actuated by signals from the effluent radiation monitors. There are redundant gas and particulate detectors in the exhaust ventilation. If the signal from any one of these four detectors exceeds a setpoint, the dampers in the ventilation system close thereby sealing the building and precluding the release of the radioactive effluent. The second exception is the liquid radioactive waste system. If the sewer discharge monitor trips, any discharge to the sanitary sewer is halted.

7.1.4 Control Console Display Instruments

The MITR console displays or otherwise provides the information needed by licensed personnel to operate the reactor. The console consists of three panels. The one to the operator's left provides information on area and effluent radiation levels. The center panel provides the position of the control devices, the reactor power level (overlapping indicators from startup to full power) and the reactor period. The panel to the operator's right shows primary flow and temperature as well as additional displays for reactor power. In addition, the console contains a centralized alarm annunciator panel that provides individual alarms from the process equipment. These are color-coded to indicate severity. Red implies a reactor scram, white is for information, and green indicates that the alarm also registers at a remote panel, which is exterior to the containment building. Each alarm is labeled with the underlying cause (e.g., low level core tank, high conductivity, low pressure compressed air). In addition, each alarm shows the corresponding procedure number so that the operator can quickly locate the appropriate response in the procedure manual.

A second function of the instrument display system is to provide essential information at panels that are outside the containment building. This information display would be of use during emergencies.

7.1.5 Radiation Monitoring System

The radiation monitoring system (RMS) consists of both area and effluent monitors. The area monitors provide indication of radiation levels throughout the containment building. These have been positioned at locations where either experimental work is performed or where work involving radioactive material is likely to be undertaken. As a result, the console operator can observe radiation levels and warn both experimenters and maintenance personnel of any unanticipated changes or hazards. The effluent monitors provide indication of the radioactivity of the air and water that leaves the building. All effluent paths are monitored. These are exhaust air (gaseous and particulate), secondary coolant (beta-gamma), and sewer discharge (beta-gamma) if in use.

All area and effluent radiation monitors are alarmed. The effluent monitors can cause system isolation as discussed in Section 7.1.3 of this report.

In addition to the area and effluent radiation monitors, the RMS includes detectors that monitor the gaseous and particulate activity of the air within the containment building. These are located in the control room, on the reactor floor, and at the reactor top.

7.1.6 Human-Machine Interface

There are two objectives to the console panel layout. The first is to provide the licensed operator with the information that is needed to monitor the MITR during both steady-state operation and the conduct of power adjustments including startup. The second is to provide information needed to identify any casualty condition or undesired trend. The first objective is met by placing information on control device position as well as the power level and period directly in front of the operator. The second is met by locating the display of both the radiation monitors and the process flow instruments immediately to the operator's left and right respectively. Other types of information, such as core tank level and reflector flow, are available in the control room. These signals tend to be constant and the operator needs to know only whether or not a setpoint is being met. If it is not met, an alarm will indicate on the annunciator panel. These instrument displays are not needed for power adjustment, nor do they provide trending information. Hence, their less prominent location is appropriate and acceptable.

7.2 Design of Instrumentation and Control System

7.2.1 Design Criteria

The MITR is a research reactor and there is no reason to continue operation during adverse conditions such as a severe natural phenomenon, a seismic event, or a fire. Administrative directions are for the reactor to be shut down should such events occur. Accordingly, while it is anticipated that the MITR's instrument and control systems would remain operable during such circumstances, once a reactor shutdown has been achieved, there is no safety need for them to do so. A shutdown reactor is protected against damage as long as the core tank integrity is maintained. The public is protected against potential radiation releases as long as the containment building integrity is preserved.

The following design criteria exist for the MITR's instrumentation and control systems:

a) Instrumentation and Control (I&C) systems are completely protected against natural phenomena such as storms by the containment building. Also, system components are securely mounted so that they should not be damaged during a safe shutdown seismic event. (Note: As discussed in Chapter 13 of this report, the building should move as a unit during an earthquake, if it moves at all.)

b) Most I&C equipment is located in the control room, which is a structure within the containment building and hence at least partially isolated from events that occur in the building. This room is equipped with a smoke detection device and fire extinguishers are in the room. Sensors and signal transmission devices are located outside the control room. Those that are important to safety are redundant and employ separate routing paths so as to avoid common mode failures.

c) Most I&C equipment is not required to operate during a credible accident. The requirement is for the reactor to be placed in a shutdown and hence safe condition and for the monitoring of certain parameters, primarily core tank level and outlet plenum temperature, to continue. Instruments that do not require electrical power, or are supplied by emergency power, are available for this purpose.

d) Both the RPS and the engineered safeguards features (ESF) actuation system automatically initiate operation to mitigate the consequences of abnormal conditions.

e) Elements of the I&C system that are important to safety include both redundancy and diversity:

   (i) The RPS has three period and three level channels with cables routed separately to avoid a common mode failure.

   (ii) The ESF actuation system for containment isolation has redundant gas and particulate (total of four) detectors.

f) The I&C systems are designed to be fail-safe whenever possible. This is usually achieved through the use of interlocks. For example, the automatic control system automatically inserts a shim blade if the regulating rod reaches its in-limit such as might occur during xenon burnout. Another example is the two-out-of-three logic that causes a scram if the period channel level signals are off-scale in any combination of high and low. However, there are circumstances where a given channel could fail and not cause a shutdown. For example, a nuclear sensor could fail low but not cause a scram if voltage were still present. This is why the nuclear safety system and other systems important to safety are both redundant and designed to avoid common-mode failures.

g) A single failure will not prevent a safe shutdown because of the aforementioned redundancy and diversity.

7.2.2 Design Basis Requirements

7.2.2.1 Reactor Control System

The Reactor Control System (RCS) has two modes: manual and automatic. The manual mode is used for both transient operation (startup, power-maneuvering, shutdown) between source range and full power (6 MW), and steady-state operation. The automatic mode is normally used only for steady-state operation at power levels in excess of 10 kW. Manual control allows the console operator to manipulate all control devices (six shim blades and the regulating rod). Under automatic control, only the regulating rod is moved. Certain safety features are designed into the system. These are that only one shim blade can be withdrawn at a time, that the shim blades can be dropped for a scram from any position of travel, that the shim blades may be run in simultaneously at normal speed, and that the console operator can take manual control at any time simply by moving the control switch for the regulating rod. There are a number of interlocks associated with the RCS:

a) <u>Startup Interlocks</u>: A reactor startup cannot be initiated unless certain prerequisites are met. This is done to ensure that operating conditions are stable prior to a startup, even though some of these conditions need not be met continuously during actual operation. The prerequisites are that all scrams are reset, that all control devices are fully inserted, that the core and reflector tanks are at overflow, and that the building differential pressure is below its setpoint. Once all conditions are met, a "reactor start" pushbutton is depressed. This bypasses all of the preceding except the scram reset requirements and allows withdrawal of the control devices.

b) <u>Subcritical Interlock</u>: This interlock applies to the shim blades. In order for any blade to be withdrawn above five inches, all blades must first be at five inches. The purpose of this interlock is to ensure that the shim blades are at a uniform bank height prior to the final approach to criticality.

c) <u>Automatic Control</u>: The MITR cannot be placed or maintained on automatic control unless certain criteria are met. These are that all shim blades be above the subcritical interlock position, that the deviation between the power setpoint and the actual power be less than 1.5%, that the regulating rod control switch be in the neutral position, and that the regulating rod be above its near-in position. Once on automatic control, failure to meet any of these conditions causes a transfer to manual control and an alarm.

d) <u>Automatic Rundown</u>: This interlock causes a shim blade to be inserted at its normal speed if the reactor is on automatic control and if the regulating rod reaches its near-in position. This scenario could occur if xenon were being burned out as the result of a startup following a brief shutdown. The concern is that if the operator did not reshim the control devices, power would rise because the automatic controller could not compensate by driving the regulating rod in any further as xenon poisoning continued to decrease.

Readouts from instruments associated with the RCS are located on the reactor console. These include indication of reactor power from a compensated ion chamber and hence accurate from the source range to full power, indication of reactor period, and a differential galvanometer that shows the deviation from a power setpoint. Also located on the console are the shim blade selector switch, control switches for both the shim blades and regulating rod, the automatic control setpoint switch, and the automatic control transfer pushbutton.

The console is also equipped with controls to shut the reactor down. These are the all-rods-in pushbutton, a minor scram pushbutton, a major scram button, the reflector dump switch, and the reflector dump air-bleed valve. The minor scram causes all blades to drop into the core. The major scram does that and it also causes building isolation and dumps a portion of the reflector. The reflector dump switch de-energizes the solenoid that keeps the dump valve closed. The reflector dump air-bleed valve interrupts the air supply that keeps the dump valve closed. Further information on these shutdown features is given in Section 7.3.2.4 of this report.

7.2.2.2 Reactor Protection System

The purpose of the RPS is to ensure that the limiting safety system settings are not exceeded as the result of transients of the type discussed in Chapter 13 of this report. The principal parameters of concern are the reactor period, the reactor power, the primary coolant flow, the coolant outlet temperature, and the core tank level. In addition to these signals, the RPS also monitors many other process system parameters such as the reflector and shield coolant flow, the status of the personnel lock gaskets, and the status of the upper grid plate's latch mechanism. A complete list is given in Section 7.3.1.1 (Table 7-1) of this report.

The setpoints at which RPS action is scheduled to activate are well below the limiting safety system settings. This conservative approach provides further assurance that a safety limit will not be approached. For example, the normal operating power level is 6.0 MW. The scram is set at 6.6 MW, the Limiting Safety System Setting is 7.4 MW, and the Safety Limit is 11.2 MW. The Safety Limit is calculated for a primary coolant outlet temperature of 55 °C, a primary coolant flow of 2000 gpm, and a primary coolant height of 4" below overflow.

The MITR can operate under either forced or natural convection. Accordingly, the RPS also has two modes. The standard mode is forced convection. All scram circuits are active and at their normal values. For natural convection heat removal, the maximum allowed power is

100 kW. Accordingly, two of the three amplifiers that provide a scram on power level are replaced with units that contain additional amplification. This allows the power level scram setpoint to be reset to a value that is consistent with the maximum allowed power for natural convection heat removal. Once these low-range amplifiers are installed, the primary flow scrams may be bypassed.

The RPS is both redundant and diverse as noted in Section 7.2.1 of this report.

7.2.2.3 Engineered Safeguards Actuation

The design basis for the ESF is that the active safeguards features be actuated both on demand and on loss of the actuating signal. For example, the building will be isolated if any one of the four plenum exhaust monitors (gas or particulate) exceeds its setpoint. The building is also isolated automatically if sample flow or electrical power is lost to any plenum monitor. Similarly, the discharge pump to the sanitary sewer trips if the signal from its associated monitor exceeds its setpoint or if the unit itself is unavailable.

7.2.2.4 Control Console Display Instruments

The design basis of the control console display instruments is that the operator be provided with the information needed to operate the facility in a safe manner. To this end, instruments are grouped as described in Section 7.1.4 of this report. Also, a central alarm annunciator panel is provided that facilitates both diagnosis and selection of the appropriate response to any abnormal condition.

A second requirement imposed on this system is that information that would be of use during an emergency must also be provided at locations exterior to the reactor building. This includes signals for core tank level, building pressure, and radiation levels within the building.

7.2.2.5 Radiation Monitoring System

The design basis of the RMS is that the console operator be provided with information on radiation levels within the building, the radioactivity concentrations of effluents, and activity in the air within the building. In addition, the RMS is designed for the automatic cessation of effluent release if abnormal activity is detected. The RMS is redundant. The sectors of the reactor building that are covered by the area monitors generally overlap so that if one unit were to fail and if a problem existed, elevated radiation levels would be observed on adjacent monitors. The effluent monitors are redundant in two modes of radiation detection. Two gaseous and two particulate detectors monitor the ventilation outlet plenum. In addition, two gaseous and two particulate detectors monitor the stack exhaust. Diversity is provided by the air monitoring detectors located within the building and by the stack area monitor.

7.2.3 System Description

A summary of the instrumentation and control system is given here with emphasis on the relation between the five subsystems. Detailed descriptions are given in Sections 7.3 - 7.7 of this report.

Figure 7-1 is a simplified block diagram that illustrates several of the more important design aspects of the reactor instrumentation and control system. The emphasis in this display is on the reactor protection, reactor control, and control console display systems with some features of the radiation monitoring and engineered safeguards actuation systems also being shown. The significance of the figure is that, as explained below, it shows that four levels of protection exist. Three of these form a protective loop about the reactor. The fourth also provides protection through building isolation.

As shown in the figure, nine separate nuclear instruments or channels are used to monitor the reactor power and period. All are located exterior to the reflector tank and hence they measure the leakage flux. Channels 1-6 provide signals to the nuclear safety system with channels 1-3 being for period and channels 4-6 for level. In addition, all six channels provide indication to the console operator. Channel 7 is the principal indicator of reactor power for the console operator. Channel 8 is a battery-operated power indication on loss of electricity, both off-site and emergency. Channel 9 supplies the automatic control circuit.

Nuclear channels 1-6 each control a relay in the withdraw permit circuit. This is a series circuit that, in addition to the nuclear channels, contains relays for each process system scram channel as well as ones for each of the startup interlock conditions. If no scram conditions exist and the startup conditions are all met, then all relays are closed and electric power can be supplied to both the protection and control systems. This is done by depressing a "reactor start" button which also has the function of bypassing the startup interlock relays.

The protection system in turn now supplies power to the electromagnets that couple the shim blades to the drives. Adjustment of the blade positions within the reactor core alters the power level and hence the signals produced by the nuclear instruments. This completes the inner loop shown in Figure 7-1. A trip of any relay (nuclear or process system) will cause a loss of power to the shim blade magnets. The magnets will therefore decouple from the drives and the shim blades will drop into the core thereby scramming the reactor. The reactor protection system is the innermost loop and it can always override the reactor control system, which forms the next loop.

The control system, which is shown in simplified form in this figure, is also enabled by the withdraw permit circuit. Once active, it provides power to the shim blade and regulating rod drives. The latter may be operated under automatic control. This is done by comparing the output of channel 9 to a specified setpoint. If the deviation exceeds an allowed band, then rod movement is initiated to adjust the reactor power. The control system forms the second loop shown in the figure.

The third loop is formed by the control console display system. Both nuclear and process system information, as well as information for the radiation monitoring system, is provided to the console operator. The operator in turn can manipulate the shim blades and regulating rod directly, or place the reactor on automatic control. Also, the operator can manually initiate a shutdown by exercising one of four options. These are to drive all blades in at their regular speed, to initiate a scram, to dump the D_2O reflector, or to both initiate a scram and simultaneously both isolate the building and dump the heavy water reflector (major scram).

A fourth loop, one that is different in its nature, is formed by the radiation monitoring system. Upon detection of a high effluent gaseous or particulate airborne activity, these systems cause the building ventilation to trip and the building to be isolated. This action isolates the reactor from the environment.

7.2.4 System Performance Analysis

Performance of the MITR instrumentation and control systems has been excellent because each component system is well-designed and well-built. Components important to safety are redundant, are tested for operability and calibrated on a regular basis, and are maintained properly. Moreover, all maintenance is documented so that any trends, such as a deteriorating part, become apparent. In addition, there is an active program to upgrade and/or replace these systems with the latest available technology.

7.3 Reactor Control System

The RCS provides for the insertion and withdrawal of the MITR's six shim blades and its fine-control regulating rod. These are the primary mechanism for the nuclear control of the MITR. The RCS consists of seven individual absorber insert/withdraw circuits, their associated interlocks, manual and automatic insertion circuitry, and an automatic control circuit.

7.3.1 Circuitry

7.3.1.1 Withdraw Permit Circuit

The withdraw permit circuit is a startup interlock that consists of a string of relays and contacts in series. Each corresponds to either a startup requirement or to a reactor scram condition. Table 7-1 lists these parts of the circuit. Each part must be closed in order for current to flow through the circuit as a whole. Hence, closure of the withdraw permit circuit means that both the

Table 7-1

Withdraw Permit Circuit Relays and Contacts

1. Reactor key switch on.
2. All absorbers fully inserted.*
3. Safety system not scrammed.
4 Channel 1 on scale.**
5. Channel 2 on scale.**
6. Channel 3 on scale.**
7. Minor scram contacts closed.
8. Medical room minor scram contacts closed (when panel key switch is on).
9. Major scram contacts closed (includes utility room and control room).
10. All-Rods-In pushbutton contacts closed.
11. Core tank at overflow.*
12. Primary coolant flow above scram point (recorder or meter).
13. MP-6 core inlet pressure above scram point.
14. MP-6A core inlet pressure above scram point.
15. ML-3 core tank level above scram point.
16. MTS-1 capillary tube temperature switch closed.
17. MTS-1A temperature switch closed.
18. Core outlet temperature recorder input below scram point.
19. Hold-down grid latched.
20. Reflector tank at overflow.*
21. Reflector flow above scram point.
22. DL-6 reflector tank level above scram point.
23. Dump valve closed.
24. PF-1 shield flow above scram point.
25. PPS-1 shield inlet pressure above scram point.
26. HF-1 secondary flow above scram point.
27. HF-1A secondary flow above scram point.
28. Inner and/or outer main airlock gasket pressure switch closed.
29. Inner and/or outer basement airlock gasket pressure switch closed.
30. Building ΔP below set point.*
31. Building overpressure switch closed.
32. Fission converter control shutter fully inserted.*

*Startup interlock.
**Two out of three must be operating.

startup interlocks and the process and instrumentation scrams are satisfied. Once the circuit is closed, current can flow both to the control blade drives (which are part of the RCS) and to the control blade magnets (which are part of the reactor protection system or RPS).

Once all of the requirements listed in Table 7-1 have been satisfied, the "reactor-start" pushbutton is depressed. This action bypasses the startup interlocks. As was explained in Section 7.2.2.1, these conditions are required to be met prior to startup. However, the startup interlocks need not be met continuously during reactor operation. All six shim blades and the fine-control regulating rod can now be withdrawn. Withdrawal of the blades at this point is limited to five inches by the subcritical interlock.

7.3.1.2 Subcritical Interlock

The purpose of the subcritical interlock is to :

a) Establish a level, below the critical position, to which the shim blades may be individually withdrawn in one step.

b) Provide a convenient reference point at which the operator can pause to make a complete instrument check before bringing the reactor to criticality.

c) Ensure that the shim blade bank is at a uniform height prior to the final approach to criticality.

This interlock can be bypassed by pulling and holding the "subcritical-bypass" lever while the shim blade control switch is manipulated. This bypass is used for special tests such as measuring individual blade drop times.

The subcritical interlock physically consists of six two-section limit switches. One switch is associated with each of the six shim blades. One section of each switch, normally closed below the subcritical interlock, opens when the associated drive reaches the subcritical position thereby interrupting current to the associated blade's drive motor. The other section of the switch, normally open below the subcritical interlock, closes when the drive mechanism reaches the subcritical position thereby forming a series circuit with the other five similar sections and the normally-open "manual-control" pushbutton. Depressing this pushbutton energizes relays whose contacts then bypass the open sections of the limit switches in the control circuit of the mechanism drive motors. Therefore, once all shim blades have been withdrawn to the subcritical position, the operator need only momentarily depress the manual-control pushbutton to continue withdrawing any given blade. Beyond the subcritical position, the operator is required by operating procedures to keep all six shim blade heights within two inches of each other. This prevents the occurrence of an unbalanced power distribution in the core. (See Section 13.2.9.1 of this report.)

7.3.1.3 Blade Withdrawal Circuits

The blade withdrawal circuits are designed to incorporate the following features:

a) Only one shim blade can be withdrawn at a time.

b) The shim blade absorbers may be dropped from their drives at any position of travel.

c) Each shim blade may be run in at its normal speed without interrupting its magnet current.

d) All six shim blades may be run in simultaneously at their normal speed without interrupting their magnet currents.

e) The fine-control regulating rod operates independently of the shim blades.

7.3.1.4 Automatic Control

The MITR has two modes of steady-state operation: manual control by the operator and automatic control of the regulating rod by a servo-mechanism. The automatic control mode cannot be initiated until certain requirements imposed by the "automatic control permit" circuit are met. These are:

a) All shim blades must be above the subcritical interlock position.

b) The deviation between the power-set and the actual power must not exceed 1.5%.

c) The regulating rod control switch must be in the neutral position.

d) The regulating rod must be withdrawn beyond its near-in position (~1.6 inches).

When the above requirements have been satisfied, the reactor can be shifted to automatic control by depressing the "automatic control" pushbutton.

When operation of the reactor is in the automatic mode, any change of status of the automatic control permit circuit will shift the reactor back into manual control and will alarm on the annunciator panel to notify the operator to take corrective action. The scram circuits directly affect all seven absorber control circuits and override all other automatic and operator actions.

Figure 7-2 illustrates the automatic control circuit. The positive direct current output of an ionization chamber is fed into a large-range, linear, direct-current amplifier. The voltage output signal from the amplifier is compared with a voltage signal from a variable power set potentiometer that is set by the console operator. If the two signals differ by about a percent, a signal is fed to the servo-control unit which then adjusts the regulating rod's position to bring the deviation back to zero. The percent deviation of the two voltage signals is displayed on both a coarse and a fine deviation meter. Whenever the reactor is on automatic control, the servo control causes the regulating rod to be automatically adjusted to hold a constant neutron flux level.

7.3.1.5 Automatic Rundown Circuit

The automatic rundown circuit is a safety circuit that is activated only during operation of the automatic control system. If, during automatic operation, the regulating rod should be driven to its near-in position by an increasing reactivity transient such as xenon burnout or decreasing coolant temperature, the automatic control circuit soon would no longer be able to keep reactor power at the specified preset level. Power would increase until a +1.5% power deviation switched control back to the manual mode. To increase the safety of the reactor and its personnel, the automatic run-down circuit activates a red light and a buzzer when the regulating rod reaches its near-in position of 1.6 inches. If the condition persists for thirty seconds, the circuit then transfers reactor control to the manual mode and drives in whichever shim blade is currently selected by the shim blade selector switch. The thirty-second delay permits the operator to correct the condition by depressing the rundown reset button, placing the reactor on manual control, and reshimming.

7.3.1.6 All-Rods-In

The all-rods-in circuit is normally used whenever it is desired to shut the reactor down completely and a scram action is not warranted. Operator insertion and withdrawal of the shim bank is normally accomplished one blade at a time through the action of the six-position shim blade selector switch and the shim blade control switch. However, this "one-at-a-time" limitation is unnecessarily time-consuming if a complete shutdown is planned. By depressing the "all-rods-in" pushbutton, the operator can simultaneously and automatically lower all six shim blades and the regulating rod. This action will cause the "withdraw permit circuit open" annunciator alarm to activate. However, in this instance, magnet current will not be lost. The shim bank insertion can be stopped by depressing the "reactor-start" pushbutton. To withdraw the blades again, the "manual-control" pushbutton must be depressed. Blade withdrawal is, as always, limited by design to one blade at a time.

7.3.2 Hardware

7.3.2.1 Blade and Rod Drive Mechanisms

See Sections 4.2.2.2 and 4.2.2.4 of this report.

7.3.2.2 Blade and Rod Position Indication

Shim blade and regulating rod positions are indicated both continuously via synchro transmitters and at certain fixed positions via switches as described below.

The seven absorber drive motors drive into gear boxes, which contain gear trains for driving coarse and fine synchro transmitters, the receivers of which give the absorber position indications on the control console. The gear boxes for the shim blade drives also contain the in-limit, out-limit, and subcritical interlock position switches that are used to operate lights on the control console. The gear box for the regulating rod differs in that it contains near-in and near-out light switches, as well as in- and out-limit light switches.

Proximity switches are activated by permanent magnets on each shim blade assembly when the blade is inserted. Two switches are located in a tube near each shim blade guide. The lower switch operates the "blade-in" light on the shim blade drive's position indicator in the control room. The upper switch is connected to the drop timer in the control room. This timer measures the interval between the initiation of a test scram signal and the 80% insertion of the appropriate shim blade.

7.3.2.3 Run-Down Relays

The run-down relay drop-out circuit consists of six current-sensitive relays, each one placed in series with an individual shim blade magnet. These relays drop out when the magnet current nears zero. If, during operation, any one of the six magnet current amplifiers should fail, or if magnet current is interrupted in any other way such as by a scram, the absorber section will fall and the mechanism of the affected shim will drive in automatically thereby assuring that the absorber is fully inserted.

7.3.2.4 Mechanisms to Decrease Reactivity

The normal means to shut the reactor down is the "all-rods-in" circuit which is described in Section 7.3.1.6 of this report.

In addition, there are several other ways by which reactivity may be decreased. These include the minor and major scram pushbuttons and the reflector dump valve. Minor scrams, which may be initiated at the reactor control room or the medical therapy room consoles, will cause all six blades to drop into the core, the blade drives to run in, and the regulating rod to run in. Major scrams, which may be initiated from the reactor control room or from the utility room, drop the blades, dump the D_2O reflector, and seal the containment building. Operation of the reflector dump valve will decrease reactivity by rapidly decreasing the level of the reflector. In addition, cycling of this valve will cause a minor scram. The dump valve is held closed by air pressure and may be opened electrically from the control room either by operating the dump valve switch or depressing the major scram pushbutton. It can also be opened mechanically by opening the emergency air bleed valve.

7.3.2.5 Nuclear Instrument for Automatic Control

The automatic control channel uses an ion chamber as its indicator of reactor power. This ion chamber senses the gamma flux and is not neutron sensitive. This is satisfactory because the gamma flux is proportional to the neutron flux for power levels of 10 kW to 6000 kW which is the range over which the automatic control system is used. Alternatively, a compensated ion chamber or a combination of a boron-lined and a standard ion chamber could be used.

7.4 Reactor Protection System

The RPS ensures that the limiting safety system settings are not exceeded. It consists of both a nuclear and a non-nuclear component.

7.4.1 Nuclear Safety System

The nuclear instrumentation system for the MITR consists of nine neutron or gamma flux monitoring channels. Each channel consists of a detector, high voltage and signal cabling, an output display device, and associated alarm, scram, or control circuitry. Channels 1 and 2 are used as startup channels, and with channel 3, have associated scram trips at a period of 10-11 seconds. Channels 4, 5, and 6 are used as power-range channels and have high flux scram trips corresponding to a reactor power level of 6.6 MW as determined by correlating the previous equilibrium value of each detector's output with the thermal power. These six instruments comprise the reactor's nuclear safety system. Channels 7 and 8 are part of the control/console display instrumentation system. Channel 7 provides a linear indication of the flux level, and channel 8 provides a flux indication if electrical power is lost. Channel 9 is part of the reactor control system. It provides a signal to the automatic control permit circuit.

All neutron detectors are mounted in either instrument ports or vertical thimbles. The detectors monitor the leakage flux in the reflector region. The degree of overlap between the channels from source range to full power operation is shown in a bar chart in Figure 7-3.

Power to all nuclear instrumentation channels except channels 3 and 8 is supplied as unregulated voltage from loads L_{21}/L_{22} via Panel 1. (See Figure 8-1). These loads are supplied by emergency power in the event of a loss of the off-site electrical power. Channels 3 and 8 are supplied by their own self-contained power supplies and do not depend on either off-site or emergency electrical power for operation.

7.4.1.1 Period Channels

Channels 1 and 2 are of the same design and are interchangeable. Each uses a fission chamber and an uncompensated ion chamber as the neutron sensing elements. The fission chamber is operated as a pulsed counter in the source range to provide visible neutron level indication at reactor startup. The uncompensated chamber is operated in the power range once the fission chamber saturates after a four decade power increase.

The output pulses from the fission chamber are amplified by a preamplifier and are then fed to a logarithmic count-rate amplifier. This amplifier first converts the pulses into a d.c. signal that is proportional to the counting rate and then amplifies the d.c. signal in a logarithmic manner. The output is indicated on a meter on the amplifier chassis and on a duplicate console-mounted meter that displays either channel 1 or 2 level as selected by the console operator. A differentiating circuit in the amplifier chassis is used to derive reactor period which is indicated on a meter on the amplifier chassis and on a duplicate console-mounted meter that displays either channel 1 or 2 period as selected by the operator. The channel output may also be recorded on a strip chart recorder that monitors channel 1, 2, or 3 as selected by the operator.

The d.c. output signal of the period-deriving network is fed to a scram amplifier as part of the nuclear safety system. Each of channels 1 and 2 has an associated safety system scram initiated by its associated scram amplifier at a period between 10 and 11 seconds.

When the fission chambers saturate at the end of their four decade range following startup, the channel is switched over to the uncompensated ion chamber. The two fission chambers are physically connected to the channel at all times and change-over is accomplished by turning down the gain on the fission chamber. This removes the fission chamber component of the signal, leaving the ion chamber, whose signal has just come on scale, as sole input to the channel. The change-over process is coordinated so that there are two operable period channels available at all times.

Channel 3 uses an uncompensated ion chamber, the output of which is fed directly to a logarithmic amplifier. The output is indicated on a meter in the amplifier chassis and on a duplicate console meter. A differentiating circuit in the amplifier chassis is used to derive reactor period with the output displayed on a meter in the amplifier chassis and on a duplicate console meter. The d.c. output of the period network is fed to a scram amplifier as part of the nuclear safety system. Channel 3 provides a safety system scram initiated by its associated scram amplifier at a period between 10 and 11 seconds. Power to channel 3 is supplied by a replaceable gel cell battery pack. A console meter displays a signal proportional to channel 3 chamber voltage.

A redundant scram has been added to the three period channels (1, 2, and 3) to ensure that the minimum number of channels monitoring reactor period will be on line and functional. The reactor will scram if two or more of the three period channels are off-scale, in any combination of high or low, simultaneously.

7.4.1.2 Level Channels

Channels 4, 5, and 6 use uncompensated boron-lined ion chambers as the neutron sensing elements. The positive d.c. current output of each channel is fed directly to an associated scram amplifier with the magnitude of the output current indicated on an edgewise meter on the front of the amplifier chassis. Each high flux trip is set to provide a reactor scram at a detector output for that channel that corresponds to a reactor power of 6.6 megawatts.

7.4.1.3 Natural Convection Operation

There are two low-range amplifiers for use when the reactor is to be operated in the natural convection mode or when certain procedures such as refuelings or blade absorber changes are performed that require work in the core tank. The low-range amplifiers allow the primary system pressure scrams on MP-6 and MP-6A to be bypassed with key switches. These amplifiers are more sensitive than the high range amplifiers because their entire range of operation is only one-twentieth that of the full-range amplifiers.

7.4.1.4 Scram Amplifier Operation

Separate scram amplifiers associated with each of the reactor period and level monitoring channels provide a scram capability for the nuclear safety system in the event of a short reactor period or a high neutron flux level. In the case of the period-monitoring channels, a derived positive d.c. signal that is proportional to inverse period is fed to the scram amplifier. In the case of the flux level channels, the positive d.c. chamber output current is fed directly to the scram amplifier.

The principle of operation of the scram amplifiers is that the positive d.c. input current is compared with a negative d.c. reference current with both the input signal and the reference current being indicated on edgewise meters on the front panel of the amplifier. The differential current produced provides a bias to control the gating operation of a field effect transistor (FET). The bias signal is positive (channel input less then reference current scram trip) during normal operation, and the FET will remain on, allowing a small-amplitude, high-frequency a.c. source signal to pass to an amplification and rectifying circuit where it is changed into a d.c. current to energize the shim blade magnets.

Each scram amplifier directly controls the power supply to its corresponding shim blade magnet (channel 1 supplies shim blade 1, etc.) and to two separate scram relays. The amount of current supplied to each magnet is controlled by a potentiometer and is displayed on an edgewise meter mounted on the face of the corresponding scram amplifier. When the positive d.c. input signal reaches the same level as the negative d.c. reference scram setting, the FET bias drops to zero thereby shutting the FET off. This stops the flow of high frequency a.c. to the d.c. rectifier network. This in turn cuts off d.c. current to the corresponding magnet, drops the associated shim blade, and de-energizes the two downstream d.c. powered scram relays. One relay cuts off all power to the paired magnet power supply (blade No. 4 for channel 1, etc.), and drops that shim blade, while the other relay opens the withdraw permit circuit thereby dropping the remaining shim blades. As a secondary action, when the withdraw permit circuit opens, the rundown relays de-energize and drop out. This causes all six shim blade drives and the regulating rod drive to be driven to their full-in positions.

7.4.2 Non-Nuclear Safety System

Table 7-2 lists the scrams associated with operation of the MITR. The instruments associated with the non-nuclear scrams are described here.

a) Low Pressure Reactor Inlet: MP-6 and MP-6A are both Bourdon tube pressure gauges that measure the primary coolant static pressure in the inlet plenum of the light-water core tank. This pressure correlates with primary coolant flow and hence MP-6 and MP-6A are protection against a loss of coolant flow.

b) Low Flow Primary Coolant: MF-1 is a flow nozzle that is mounted in a straight section of the primary piping. A strain gauge circuit that is coupled to the differential pressure transmitter's diaphragm provides a signal proportional to the pressure drop across the nozzle. The flow rate is proportional to the square root of this pressure drop (Bernoulli equation). Both the MF-1 and the MP-6/6A scrams are set above the LSSS of 1800 gpm.

c) Low Level Core Tank: ML-3 is a float switch that is mounted within the core tank. ML-3 causes a scram if the coolant level drops four inches below overflow.

d) High Temperature Reactor Outlet: MTS-1 is a capillary tube-type thermoswitch. MTS-1A is a thermocouple. Both measure the temperature of the coolant as it exits the core tank. For MTS-1, changes in coolant temperature will cause the silicone-based fluid that fills the capillary tube and an associated coil to expand and move as indicated.

e) High Temperature Reactor Outlet Recorder: MT-5A is a platinum resistance temperature detector (RTD). It is mounted in the primary piping at the suction of the primary coolant pumps. MT-5A, MTS-1, and MTS-1A are all set to scram below the LSSS of 60° C.

f) Low Flow D_2O: DF-1 is an orifice plate that generates a signal using the same principal as MF-1. DF-1 causes a scram above the minimum required value of 75 gpm.

g) Low Level D_2O Reflector: DL-6 is an insulated stainless steel probe that is mounted in the reflector's overflow line. DL-6 causes a scram if the reflector level drops four inches below overflow.

h) Low Flow Secondary: HF-1 and HF-1A are orifice plates that generate signals using the same principal as MF-1. Both HF-1 and HF-1A scram on low flow. Neither scram is required by the technical specifications.

i) Low Flow Shield Coolant: PF-1 is an orifice plate that generates a signal using the same principal as MF-1.

j) Low Pressure Shield Inlet: PPS-1 is a Bourdon-tube pressure switch that measures the pressure at the discharge of the shield coolant pump. This pressure corresponds to the shield flow. Both PF-1 and PPS-1 cause a scram above the minimum required value of 50 gpm.

k) Building Overpressure: XPS-3 is a pressure switch which measures the differential pressure between the reactor control room and the atmosphere. It causes a scram below the maximum allowed value of 3 inches of water.

l) Hold-Down Grid Unlatched: In order to latch the grid, it is necessary to reset a mechanical switch. If the grid becomes unlatched, the switch opens and causes a scram.

7.5 Engineered Safeguards Actuation

This system includes passive, manual, and active components. The passive ones (natural circulation valves, anti-siphon valves, and building vacuum breakers) and the manual ones (building pressure relief and emergency core cooling initiation) are described in Chapter 6 of this report. The active components (blockage of effluent release path) are discussed in Section 7.7 of this report.

Table 7-2

List of Reactor Scrams

Scram Condition	Scram Setpoint
Withdraw Permit Circuit Open*	Minor, major, or medical facility scram pushbutton depressed Key switch off Dump valve open
Nuclear Safety System Scram	
Period	≥ 7 sec
Power	≤ 6.6 MW
Period Channel Level Signal Off-Scale	Less than two period channel level signals on scale
Low Pressure Reactor Inlet, MP-6, MP-6A	Pressure corresponding to > 1800 gpm
Low Flow Primary Coolant, MF-1	≥ 1800 gpm
Low Level Core Tank, ML-3	≥ -4 inches
High Temperature Reactor Outlet, MTS-1, MTS-1A	≤ 60° C
High Temperature Reactor Outlet Recorder, MT-5A	≤ 60° C
Low Flow D_2O, DF-1	≥ 75 gpm
Low Level D_2O Reflector, DL-6	≥ -4 inches
Low Flow Secondary, HF-1, HF-1A	≥ 400 gpm
Low Flow Shield Coolant, PF-1	≥ 50 gpm
Low Pressure Shield Inlet, PPS-1	Pressure corresponding to ≥ 50 gpm
Building Overpressure, XPS-3	≤ 3" water above atmospheric
Main Personnel Lock Gaskets Deflated	Both gaskets deflated
Basement Personnel Lock Gaskets Deflated	Both gaskets deflated
Hold-Down Grid Unlatched	Grid unlatched
Low Flow Fission Converter	≥ 45 gpm (forced circulation mode only)
Low Level Fission Converter	≥ 2.1 meters (forced circulation mode) ≥2.5 meters (natural convection mode)
High Power Fission Converter	≤ 20 kW (natural convection mode only)

* The withdraw permit circuit open alarm activates on all scrams. For the conditions listed here it will be the only indication of the scram.

7.6 Control Console Display Instruments

Table 7-3 lists the instruments that read out or otherwise indicate at the reactor console. Many additional instruments indicate elsewhere in the control room.

7.6.1 Linear Flux Channel

Channel 7 utilizes a compensated ion chamber as its neutron sensing element. This channel is the console operator's principal indication of reactor power. The output of the ionization chamber is fed to a modified digital picoammeter unit. The picoammeter is a low-level (d.c.) amplifier with seven log current ranges (seven decades) that may be selected with pushbutton switches on the front panel. The entire operating range of the reactor can be followed on the console-mounted display by adjusting the range of the picoammeter during reactor startup and shutdown.

There are two outputs taken from the picoammeter. One is a d.c. signal of 0 to 1 volt which provides current to drive a console-mounted differential galvanometer through a range switching network. A bucking voltage, consisting of a dry-cell battery and an adjustable voltage divider, is connected in opposition to the voltage from the picoammeter so that the deviation from a desired setpoint may be observed on the galvanometer. The range switching network includes shunts that allow the full-scale deflection of the galvanometer to correspond to 1%, 10%, or 100% of the full-scale meter reading of the picoammeter. The other output is a d.c. signal fed to a linear strip chart recorder mounted on the console.

7.6.2 Channel 8

This channel utilizes an uncompensated ion chamber as its neutron sensing element. Its principal function is to provide an indication of the reactor power level when off-site electrical power has been lost. In addition, it provides a backup to the linear flux channel when the reactor is above a few kilowatts and it provides an alarm if the reactor power exceeds its setpoint.

Table 7-3

Console Instruments

A. Left Section

1. Left annunciator panel alarms
2. Area radiation monitors
3. Effluent monitor flow indicator lights
4. Effluent radiation monitors (plenum gas, plenum particulate, stack gas, stack particulate, secondary coolant, core purge, and sewer pump discharge)
5. Radiation monitor recorder
6. Closed-circuit television camera monitor
7. Intercom master panel

B. Center Section

1. Center annunciator panel alarms
2. Channel 1, 2, or 3 recorder
3. Channel 7 recorder
4. Core outlet temperature MT-5A recorder
5. Shim blade and regulating rod position indicators
6. Channel 8 meter
7. Channel 1 or 2 level and period meters
8. Differential galvanometer
9. Automatic control deviation meters
10. Channel 3 level and period meters
11. Clock

C. Right Section

1. Right annunciator panel alarms
2. Primary flow, ΔT, and thermal power recorder
3. Thermal power indicator
4. Core outlet temperature MTS-1A digital indicator
5. Core outlet temperature MT-5A digital indicator
6. Nuclear instrument channel displays.

All power required by channel 8 is supplied by a battery mounted in the back of the control console in the control room. A trickle charger, which keeps a constant voltage across the battery, maintains the battery at full charge. The battery level for channel 8 can be read on the rear of the console. The high voltage required to operate this chamber is supplied by means of a d.c. to d.c. voltage converter. The output current of the chamber is fed to a linear scale microammeter and to a meter relay. The microammeter provides power indication above a 500 kW threshold level. A signal taken from this meter is fed to a log range digital picoammeter. This picoammeter has seven pushbutton-controlled range switches that provide a digital output indication from 0.1-100% power.

The meter relay is used to provide a high level alarm that is set at a power level above the operating point (6.0 MW) and below the scram setpoint (6.6 MW) by comparing the previous channel 8 equilibrium value with the equilibrium thermal power.

7.6.3 Thermal Power Indicator

Thermal power is estimated by a digital-hybrid device that first takes the product of primary flow and the core ΔT and then adds in a factor to account for heat removed by the reflector and shield systems. The factor was obtained from the MITR's previous operating data. The thermal power equals the actual (neutronic) power if the reactor is at equilibrium. The MITR has substantial heat capacity, especially in the graphite reflector and shield. Hence, thermal power often lags neutronic power. Thermal power is used to calibrate the nuclear instruments once the reactor is at equilibrium. Administrative procedures require that the thermal power be calculated, recorded, and compared to the nuclear instruments at least weekly.

7.6.4 Annunciator Alarm System

An annunciator alarm panel that is positioned directly over the console in the control room displays the status of the continuous surveillance alarm system. The main annunciator panel consists of sixty-four modular plug-in relay assemblies. Each unit has a backlighted identification plate. Alarms which are accompanied by reactor scrams have red name plates while informational alarms are equipped with white or green name plates. Some of the relay assemblies are twin-point units which will indicate two alarms conditions. A separate alarm condition is engraved on each half of these name plates and the unit is divided such that the proper half of the plate will illuminate when the alarm is actuated. The annunciator panel has three sections: a central panel of forty-four units and two side panels of ten each.

There are two pushbuttons on the console that are used to acknowledge and reset an alarm thereby maintaining the vigilance of the system for subsequent alarms. A third pushbutton is used to test the operation of the bell and all alarm lamps.

When an alarm is energized, a bell sounds and a light behind the appropriate translucent name plate flashes brightly on and off thereby indicating which alarm module is annunciating. The operator depresses the "annunciator acknowledge" pushbutton. This action silences the bell but leaves the alarm module's name plate both brightly and continually lit. When the alarm condition has been corrected, the bell again sounds and the alarm plate flashes dimly. The operator then depresses the "annunciator reset" pushbutton to silence the bell and extinguish the light.

Operation of the twin-point units is only slightly different. When the alarm is energized, the operator depresses the "annunciator reset" pushbutton. This action silences the bell and leaves the alarm module's name plate both brightly and continually lit. The twin-point units do not require reset action by the operator and do not flash or ring the bell when the alarm condition clears.

7.6.5 Weekend Alarm System

7.6.6 Remote Instrumentation

Table 7-4

MITR Remote Instrumentation

7.7 Radiation Monitoring System

The radiation monitoring system consists of twelve effluent monitors including the one in the core purge system, and ten area monitors including the ones in the ventilation exhaust stack and in the secondary chemistry shielded enclosure. The system ensures that the release of radioactive effluents to the environment is within the limits of MITR Technical Specifications which provide that, with certain very specific exemptions, any release of radioactive effluents will conform to 10 CFR Part 20, the Code of Federal Regulations.

The output of each effluent and area monitor is displayed in the control room. Any monitor will initiate an alarm on the annunciator panel if its output exceeds a preset trip level. In addition, if the trip level of any one of the plenum particulate or gas monitors is exceeded, the ventilation system will be secured and the intake and exhaust dampers automatically closed thereby ensuring containment isolation. One each of the gas and particulate channels from the plenum and stack monitors are continuously recorded on a strip chart recorder which serves as one of several permanent records for airborne effluent releases.

7.7.1 Area Monitors

Surveillance of the radiation levels within the containment and in the ventilation exhaust stack is performed by the area monitors listed in Table 7-5. A minimum of one area radiation monitor, capable of warning personnel on the reactor floor of high radiation levels, is required to be operating whenever the reactor floor is occupied. The area monitors, which were built in 1998 by Ludlum Measurements, Inc., consist of detector signal conditioners (Model No. 306-15), control units (Model No. 306-3), and remote units (Model No. 307-5).

Table 7-5

Area Monitors

Monitor	**Range**	**Remote Indicator**
Reactor Floor #1	0.1 to 10K mrem/hour	
Reactor Floor #2	0.1 to 10K mrem/hour	
Reactor Floor #3	0.1 to 10K mrem/hour	
Reactor Top	0.1 to 10K mrem/hour	
Primary Chemistry	0.1 to 10K mrem/hour	
Secondary Chemistry	0.1 to 10K mrem/hour	
Medical Room	0.1 to 10K mrem/hour	Yes
Secondary Chemistry Shielded Cell	0.1 to 10K mrem/hour	Yes
Spent Fuel Pool	0.1 to 10K mrem/hour	
Stack	0.1 to 10K mrem/hour	Yes

7.7.1.1 Detection System

The detection system for each of the area monitors is referred to as a Detector Signal Conditioner (DSC). Each DSC contains an energy-compensated Geiger-Mueller tube, a high voltage circuit, a stage pulse amplifier, a comparator, and a differential line driver. Two cables are used to power, control, and direct detector signals to and from each DSC. Each DSC has its own dedicated control unit.

The output of each area monitor is displayed at the control unit and locally on the DSC, except for the stack, medical room, and secondary chemistry shielded cell monitors which display at the control unit and on remote indicators. The stack area monitor, which is located 40 feet up the 150 foot ventilation exhaust stack, is inaccessible during reactor operation and is provided with a remote indication in both the control and utility rooms. The basement medical therapy room unit provides a remote indication of radiation levels within the room on a meter near the medical control panel and the room's viewing window. (Note: This unit is distinct from the one listed in Chapter 17 of this report, which is required by the MITR BNCT program's technical specification.) The secondary chemistry shielded cell monitor detects radiation levels inside the cell itself. It also provides a signal to the interlock that controls the automatic transfer of samples to laboratories outside the containment building. The DSC is located inside the enclosure and uses a remote meter mounted on an outer wall to provide local indication.

The DSC chassis and remote indicators contain LED displays which provide local indication of the area radiation level in mrem/hour along with alarm, overrange, and detector failure status lights. (Note: While many radiation monitors including these have displays in mR/hour, they can no longer be described that way, because the milli-Roentgen (mR) is no longer a recognized unit. It is acceptable to interpret the displays as millirem (mrem) per hour.)

7.7.1.2 Control Units

The control units that power each DSC provide control room indication and process alarms, and control all self-testing of each area monitor. Each unit is powered separately by 120V a.c. and distributes power to its attached DSC and its remote indicator (if used). The radiation level is presented using an LED display with a digit span of 0.0 to 9999.9. The display is updated every two seconds with the same signal fed to a parallel analog meter with a 5 decade logarithmic scale.

The "overrange" LED illuminates whenever the DSC is in an overrange condition normally associated with very high radiation fields. The digital display will indicate an overrange count rate by displaying all "9s". Detector saturation is indicated by the display of all dashes ("–"). Each control unit also outputs its display to a radiation recorder for trend analysis.

Functional testing of a DSC is provided by a remotely activated check source. When the "source test" pushbutton is depressed on a DSC's control unit, an internal motor moves a Cs-137 check source close to the detector. When the "source test" pushbutton is released, the check source retracts behind a shield.

Each control unit contains two latching alarm bistables. One is for high radiation levels, the other for a detector failure. The "high alarm" bistable operates in conjunction with an alarm set thumbwheel switch on the front panel. The area monitor's signal is compared to the alarm setpoint every half second. If the actual radiation level is equal to the trip level, the bistable will de-energize causing an annunciator panel high radiation area monitor alarm and the control unit's front panel "alarm" LED to illuminate. The "detector fail" bistable is controlled by the pulses received from the DSC. If no pulses are received over a selected interval, the bistable de-energizes causing the "Det Fail" LED to illuminate on the front panel along with an annunciator panel trouble radiation area monitor alarm.

If either bistable de-energizes, the following alarms will be activated:

- Control unit "Alarm" or "Det Fail" LED
- Annunciator panel "High Radiation Area Monitor" or "Trouble Radiation Area Monitors" alarm
- DSC (or remote indicator) audible alarm
- DSC (or remote indicator) "Alarm" or "Det Fail" LED

The DSC audible alarm is silenced by depressing the "Audio Acknowledge" pushbutton on the control unit. When the radiation level decreases below the alarm setpoint or pulses are detected, the alarm can be reset. Both bistables will de-energize if the control unit is turned off or removed for maintenance. If this occurs, bypasses may be installed to restore annunciator alarm capability to the other area monitors.

7.7.2 Effluent Monitors

The effluent monitors continuously detect activities within the gaseous and liquid effluent streams. Their purposes are to measure the concentration of radioactive effluents, to warn of abnormal activity levels and, in the case of the plenum and sewer monitors, to take preventive action in the event of such abnormal levels. The effluent monitors are listed in Table 7-6.

Although the detector types differ depending on the nature of the effluent being monitored, their electronic configurations are similar. Specifically, a pulse signal, whose average count rate is proportional to the monitored activity, is fed to a solid state ratemeter in the control room where the information is presented on a logarithmic scale. Each ratemeter display has a range of 20 cpm to 200,000 cpm.

The panel meter also serves as an adjustable-trip meter relay. When the meter reading exceeds the high trip-point setting, an alarm relay is activated causing a high level radiation alarm on the annunciator panel. This action also energizes a red light on the ratemeter front panel which will stay lit until the trip-point is no longer exceeded and a reset button on the same panel is depressed. The meter also has a low level trip which activates the amber light on the front panel and a trouble radiation monitor alarm on the annunciator panel.

Table 7-6
Effluent Monitors

Monitor	Detector Location	Detector Type
Secondary Water Monitor 1	Equipment Room	Scintillation
Secondary Water Monitor 2	Equipment Room	Scintillation
Sewer Monitor	Back Stairwell	Scintillation
Plenum Gas 1	Equipment Room	G-M pancake-type 1.4mg/cm^2 window
Plenum Gas 2	Equipment Room	G-M pancake-type 1.4mg/cm^2 window
Plenum Particulate 1	Equipment Room	G-M pancake-type 1.4mg/cm^2 window
Plenum Particulate 2	Equipment Room	G-M pancake-type 1.4mg/cm^2 window
Stack Gas 1	Base of Stack	G-M pancake-type 1.4mg/cm^2 window
Stack Gas 2	Base of Stack	G-M pancake-type 1.4mg/cm^2 window
Stack Particulate 1	Base of Stack	G-M pancake-type 1.4mg/cm^2 window
Stack Particulate 2	Base of Stack	G-M pancake-type 1.4mg/cm^2 window
Core Purge Monitor	Utility Shelf	G-M pancake-type 1.4mg/cm^2 window behind a stainless-steel window

7.7.2.1 Ventilation Monitors

Monitors that sample the ventilation exhaust gases are located in the equipment room and at the base of the stack. Two gaseous and two particulate monitors are installed at each location. Those located in the equipment room are referred to as "plenum" effluent monitors while those in the stack base are termed "stack" effluent monitors. (These should not be confused with the stack area monitor which is located forty feet up the stack and has separate circuitry as described in Sections 7.7.1 and 7.7.1.1 of this report.)

The detector for each monitor is a two-inch diameter pancake-type GM detector placed in a lead shield to reduce background. In addition to the detector, the particulate monitors contain a filter paper assembly with both HV-70 paper and activated charcoal filters.

Each set of particulate and gaseous monitors has its own blower and instrumentation. If flow is lost through a monitor set, a trouble radiation monitor alarm will activate. In addition, if flow has been lost through a plenum set, the ventilation system will be automatically isolated. A radiation monitor that continually samples the effluent air stream and is capable of automatically securing the building ventilation ducts is required to be operating whenever containment integrity is needed. Also required under the same condition is a radiation monitor (gaseous, particulate, or area) in the exhaust stack.

a) Plenum Monitors: Redundant plenum gaseous and particulate monitors, located in the equipment room, continually sample the effluent air at the upstream end of the exhaust holdup plenum. A high level alarm on any one of these four plenum monitors will, besides energizing the high level radiation monitor alarm on the annunciator panel, cause the building ventilation system's intake and exhaust fans to stop and the isolation dampers to close before the sampled gas can pass by the exhaust damper. This ensures that whatever the abnormal activity detected, it will be trapped within the reactor building.

   In the event of a failure of the main dampers to shut within less than a specified time from the receipt of a close signal from the monitors, the auxiliary backup dampers will automatically close. The auxiliary dampers are weighted, hinged, steel-flap, gravity drop dampers. In addition, both sets of dampers can be closed manually by the operator.

b) Stack Monitors: The stack monitor sampling system is identical to that of the plenum monitors. These monitors determine whether any high level release has actually taken place in the event of a ventilation trip.

The redundant stack gaseous and particulate monitors have electronic components similar to the other effluent monitors.

7.7.2.2 Secondary Water Monitors

Two redundant water monitors are positioned in a shielded location. Each system separately samples water from the secondary coolant pipe leading to the cooling towers. (See Figure 5-4.) Sample water is returned to the main piping. Flow through the monitors depends on the pressure differential produced by the main pumps in the secondary system. Activity could be introduced to the secondary water if a leak developed within any of the heat exchangers serviced by the secondary water system. A D_2O heat exchanger leak would be of particular concern because of possible high tritium activity in the heavy water.

Each system uses a gamma-sensitive scintillation detector which views a volume of water contained in a lead shield. It cannot detect tritium but is sensitive to N-16 and F-18 which are present in both the heavy-water reflector and the light-water coolant whenever the reactor is operating. The N-16 isotope has a half-life of a few seconds while that of F-18 is approximately fifteen minutes. Hence, once the reactor is shut down, these isotopes rapidly decay and the water monitors are no longer capable of detecting any leakage. Normally a flow of at least two gpm is maintained through each lead shield as indicated by local flow meters. Flow switches will produce a trouble radiation monitor alarm if the flow falls below one gpm. (Refer to Section 7.7.2.5.)

Sampling of secondary water for tritium is performed at least once every twenty-four hours during those periods when the reactor is in normal operation. This allows detection of very small leaks. If the secondary water tritium concentration equals or exceeds 1 μCi/liter, the reflector heat exchanger is to be isolated until tritium leakage into the secondary system has been controlled.

7.7.2.3 Sewer Monitor

A detector system, similar to the water monitors, is used to monitor the liquid discharge from the waste storage tanks to the sanitary sewer. When the waste tanks are to be discharged, the liquid waste is thoroughly mixed in order to obtain representative samples for counting. If both the radiation levels and the activity to be discharged are within permissible limits, the liquid can then be pumped to the sewer. A high level trip on the sewer monitor will stop the sewer pump and energize the high level radiation monitor alarm on the annunciator panel.

Discharges from the waste tanks are necessary only several times per year. Hence, the sewer monitor may normally be used for another purpose. Specifically, it usually is used to monitor the waste water that is intermittently pumped from the sump to the waste tanks. When used in this mode, a high level trip on the sewer monitor will stop the sump pump and activate the high radiation monitor alarm on the annunciator panel.

7.7.2.4 Core Purge Monitor

The core purge monitor uses a pancake-type GM detector which monitors activity in the core purge gas flow through a stainless steel window. A blower located in the equipment room draws reactor building air through an inlet filter, over the water in the main core tank, and past the detector. It is discharged through an outlet filter to the main ventilation exhaust duct. The point at which the core purge flow merges with the building exhaust ventilation is located upstream of the plenum effluent monitors.

The monitor has a preset alarm which, in addition to energizing the high radiation core purge alarm, will automatically turn the core purge blower off and isolate the core purge system. This function allows building ventilation to be maintained even if high activities are present in the air over the main core tank. However the reverse is not possible because the core purge blower is interlocked with the building ventilation so that if ventilation is secured, the blower automatically stops.

Additional information on the core purge monitor is given in Section 5.2.1.7 of this report.

7.7.2.5 Trouble Radiation Monitor Alarm

The purpose of the "Trouble Radiation Monitor" alarm is to ensure that the operator is cognizant of the loss of a radiation monitor channel. Table 7-7 lists conditions which will produce this alarm. Loss of flow through the plenum monitoring systems will also result in reactor containment building isolation.

The plenum monitoring system is equipped with a key switch that allows one plenum blower to be bypassed at a time. This feature permits the blower filters to be changed at power without tripping the ventilation.

7.7.3 Other Radiation Monitors

7.7.3.1 Reactor Floor Argon-41 Monitor

The Ar-41 monitor on the reactor floor consists of a 30 mg/cm^2 metal wall, beta-sensitive Geiger tube in a one-liter chamber that is surrounded by four inches of lead. The air in the reactor building is sampled continuously inside this chamber by air flow through a system that penetrates the detector shielding. This air is prefiltered using HV-70 filter paper to eliminate particulates. The detector will respond under normal operating conditions to Ar-41. The resulting signal is fed to a recording count rate meter that has a range setting of 0 - 1000 cpm and is located next to the chamber.

7.7.3.2 Control Room Argon-41 Monitor

An Ar-41 monitor of the same design as that used on the reactor floor (Section 7.7.3.1 of this report) is located in the control room.

Table 7-7

Trouble Radiation Monitor Alarm

Cause	Action	Bypass Switch
Low level on any effluent monitor	Alarm	No
Any effluent monitor not set in CPM mode	Alarm	No
Loss of flow to either secondary water monitor	Alarm	No
Loss of flow to any plenum monitor	Alarm, ventilation trip, and dampers close	Yes*
Loss of flow to any stack effluent monitor	Alarm	No

* Only one blower may be bypassed at a time. One blower provides flow to one set of plenum particulate and gaseous monitors.

7.7.3.3 Reactor Top Air Particulate Monitor

The particulate monitor on the reactor top consists of a 30 mg/cm^2 metal wall, beta-sensitive Geiger tube in a volume shielded by four inches of lead. The air in the reactor building is sampled continuously into this chamber by air flow through a system that penetrate the detector shielding. The air flow is through a filter paper to collect any airborne particulates. The filter paper and holder are in close proximity to the sensitive window of the G-M detector permitting the detection of airborne particulates collected on the filter paper. The resulting signal is then fed to a recording count rate meter that has a range setting of 0 - 1000 cpm and is located next to the chamber. The same signal is also fed to a remote meter that is located outside the containment building in the Reactor Operations Office.

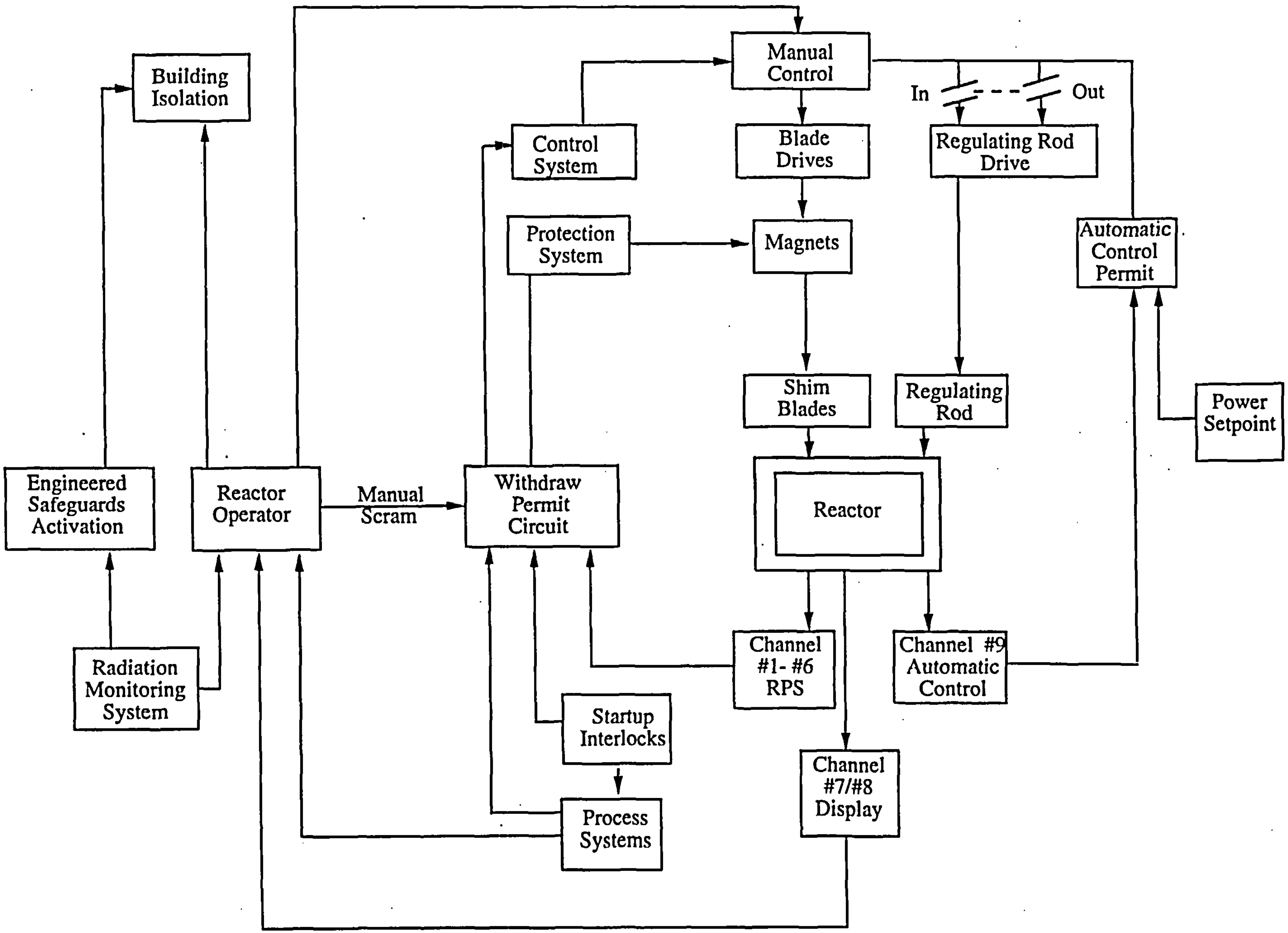

Figure 7-1. Simplified Diagram of Reactor Instrumentation and Control Systems

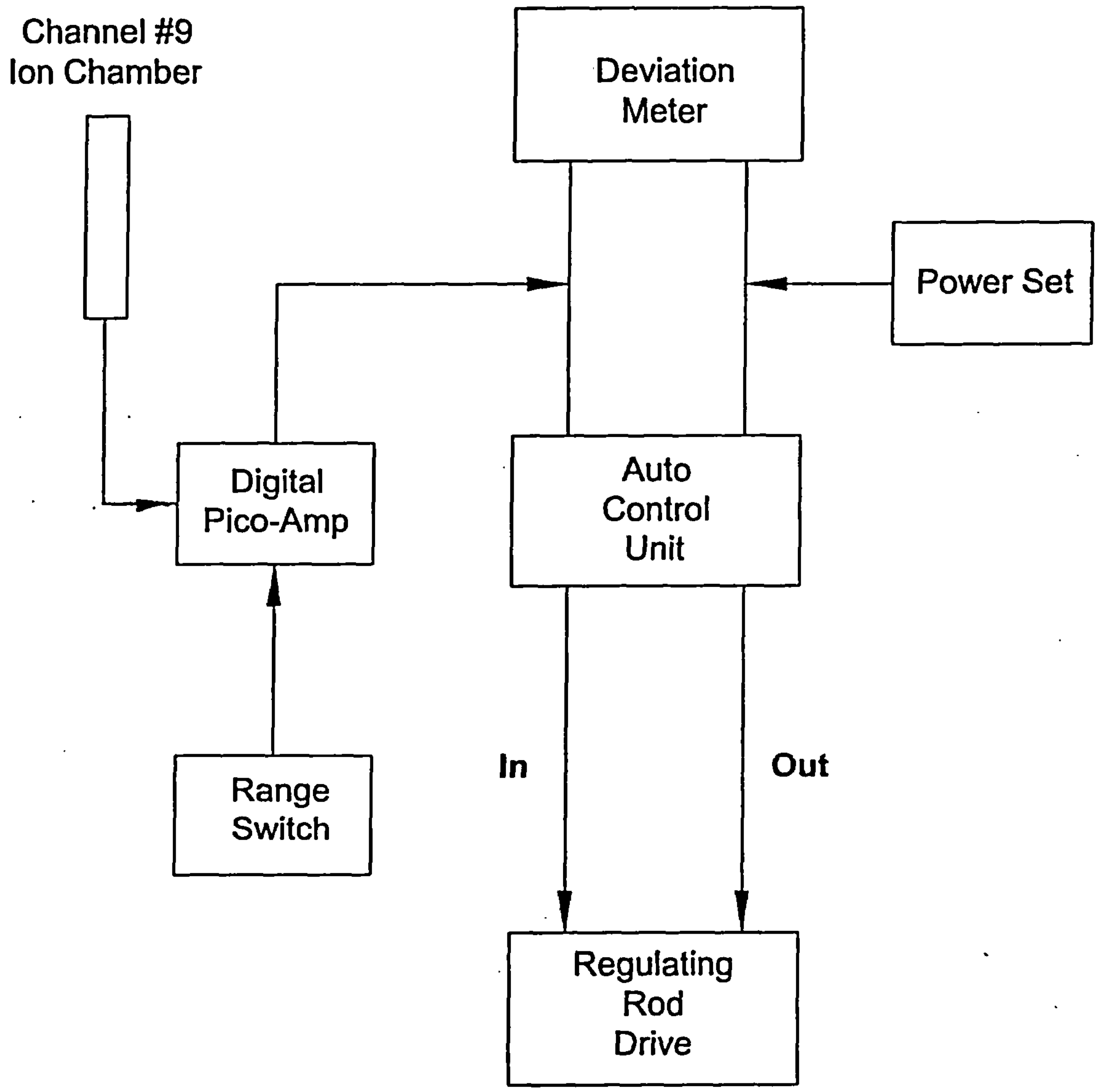

Figure 7-2
Functional Block Diagram of Automatic Control Circuit

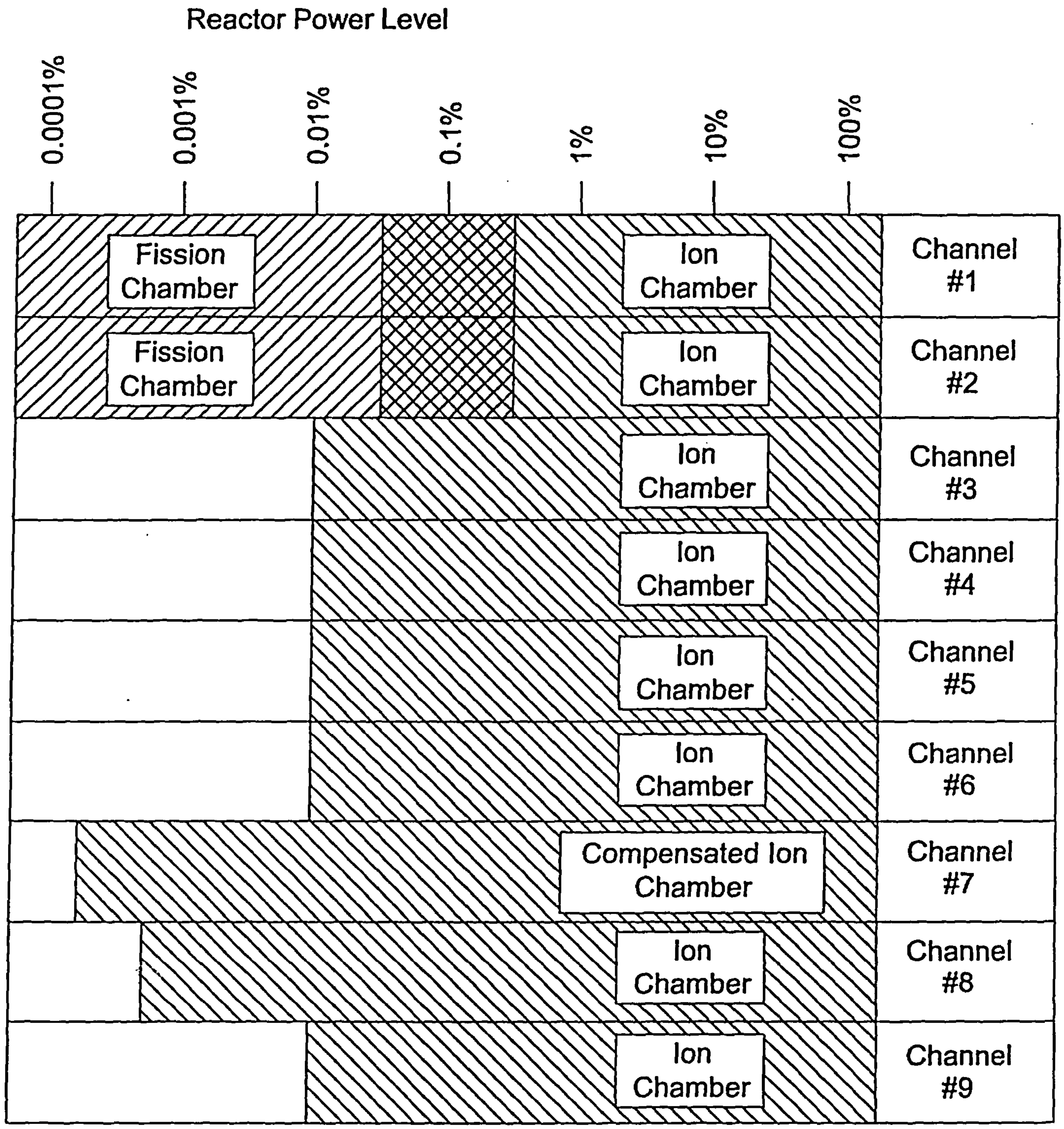

Figure 7-3
Channel Utilization Graph

Chapter 8

Electrical Power Systems

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Chapter 8

Electrical Power Systems

8.1 Normal Electrical Power Distribution

8.1.1 Design Basis

The functional requirements of the MITR's normal electrical power distribution system are: 1) to supply electrical power for all motors, pumps, and instrumentation associated with the operation of the reactor and fission converter; and 2) to facilitate a safe reactor shutdown if the off-site power supply is interrupted. Relative to the latter, the following passive safety features exist. In the event of loss of normal electrical power:

a) The six reactor shim blades, which are held up by electromagnets, will drop into the core because of gravity. This shuts the reactor down.

b) The D_2O dump valve, which is held closed by air pressure that is in turn controlled by a solenoid valve, will open. This dumps the reflector and further shuts the reactor down.

c) The isolation dampers in the containment building ventilation system close, thereby isolating the containment and precluding any potential release of radioactive material.

d) Natural convection, as discussed in Chapters 4 and 5 of this report, will remove decay heat in the case of a long-term power outage.

In addition to the above, certain electrical loads will be transferred automatically to emergency power. These provide for both lighting and continued monitoring of the reactor. Finally, auxiliary pump MM-2 can be restarted on emergency power and used to provide forced convection cooling for decay heat removal as described in Chapter 5 under Mode 1 - Emergency Cooling.

8.1.2 System Description

Figure 8-1 is a schematic of the MIT Research Reactor's normal and emergency power distribution. Two incoming 13.8 kV power lines pass underground from Albany Street to the MITR's utility room. These lines feed into separate, main, air-circuit breakers set to trip at 1200 amperes. Normally only one circuit breaker is closed at a time. In the event of loss of off-site power on the selected feeder, the system can be switched to the other by means of the 1200 ampere circuit breakers. Electric utility personnel are the only individuals allowed to perform this transfer. An ammeter, an ammeter selector switch, the main disconnect control switch, and four time-delayed, over-current relays are located on the front panel of each breaker housing. The main feeder runs from either circuit breaker through the metering compartment to a 750 kVA transformer. A power factor correction capacitor bank is used to balance inductive and capacitive loads automatically. Six 480 volt lines from this transformer supply power to:

a) Two additional step-down transformers (described below).

b) Motor control centers one and two (located in the utility and equipment rooms).

c) Reactor Laboratory Building (NW12).

d) Experiment power panels (located in the reactor containment building).

Major items of equipment energized from the two motor control centers are listed in Table 8-1.

A 200 ampere breaker supplies a 150 kVA (208/120 volt), three-phase, air-cooled, step-down isolation transformer. This transformer, which is located exterior to the reactor containment building in the utility room, in turn supplies three loads: the waste tank shed, the lighting panel in the utility room, and Panel No. 2.

e) Waste Tank Shed: This circuit supplies power to the liquid radioactive waste storage tank shed.

f) Utility Room Lighting Panel: This panel supplies power to the utility room lights, the reactor air compressor, the building cathodic protection system, the charger for the emergency battery, and outside lighting.

g) Panel No. 2: This panel, which is the main 208/120 volt distribution center for the reactor containment building, supplies both the normal and emergency lighting panels as well as Panel No. 2A.

Table 8-1

Equipment Energized at Motor Control Centers No. 1 and No. 2

MCC-1 (Utility Room)	MCC-2 (Equipment Room)
1. Main Personnel Lock Hydraulic Pump	1. Recombiner Blower GM-1
2. Sewer Pump	2 Polar Crane
3. Intake Damper Hydraulic Pump	3. Hydraulic Lift - Elevator
4. Exhaust Fan	4. Hydraulic Pump - Truck Lock Rams
5. Intake Fan	5. Truck Lock Door Lifts
6. Cooling Tower Fans	6. Medical Door Drive
7. Control Circuit Transformer	7. Sump Pump
	8. Exhaust Damper Hydraulic Pump
	9. Pneumatic Blower
	10. Primary Pump MM-1
	11. Primary Pump MM-1A
	12. Secondary Pump HM-1
	13. Secondary Pump HM-1A
	14. Basement Auxiliary Fans (3)
	15. Water Shutter Pump BM-1
	16. D_2O Transfer Pump DM-2
	17. D_2O Pump DM-1
	18. Control Circuit Transformer
	19. Shield Pump PM-1

A 100 ampere breaker supplies a 45 kVA (208/120 volt), three-phase, air-cooled, step-down isolation transformer that is also located in the utility room. It supplies the Experimental Receptacle Load Center, which is located in the equipment room, as well as Panel No. 1/1A, which is located behind the reactor console in the control room. Panel No. 1/1A is a distribution center for all reactor instrumentation and control systems. Certain of its circuits are transferred automatically to emergency power in the event of loss of the normal power supply. Table 8-2 lists loads on Panels No. 1 and 1A and indicates those that transfer to the emergency bus.

Both the 150 and the 45 kVA transformers are connected in the "star" configuration. The input is 480 volts (line - line). The output is 208 volts, 3ϕ and 120 volts to neutral.

Electric power is transferred into the reactor containment building via conduits that penetrate the steel shell of the reactor building through seven pothead seals in the utility room. The conduits pass down through the concrete wall of the reactor building and into the equipment room. Additional control and communication wires penetrate the containment shell through terminal boxes mounted in the utility room.

All electrical equipment is connected to a common ground system. The ground consists of bare soft-drawn copper wires that run through the utility and equipment rooms and are connected to the steel shell of the building at a single point.

All electrical equipment associated with the MITR is free of PCBs. The system conformed to the appropriate electrical codes when it was originally built. All additions/modifications to the system circuits' original construction were done in accordance with the code requirements that were applicable at the time of the additions. None of the system has been exempted from code requirements. All equipment is periodically inspected, cleaned, serviced, and repaired/upgraded, if necessary, by MIT's Department of Facilities.

8.1.3 Electrical Power Capability

Normal power requirements for the reactor and fission converter are 325 amperes at 480 volts, 60 Hz a.c. This equates to about 270 kWe. The system is capable of supplying 750 kWe.

Table 8-2

Equipment and Circuits Energized from Panels 1 and 1A

Panel No. 1	Panel No. 1A
1. Startup Channels	1. Equipment Room 208V Outlets
2. Spare	2 PCCL Charging Pump
3. Spare	3. Spare
4. Equipment Room Drop Light	4. Sensor/IASCC Coolant Pump Back-Up
5. Spare	5. Stack Base 208V Outlets
6. Recombiner Heaters	6. Auxiliary Pump MM-2
7. Misc. D_2O Instrumentation	7. Spare
8. Spare	8. Spare
9. Medical Facility Lighting	
10. Medical Therapy Control Panel	
11. Medical Therapy Lighting	
12. Blowdown Controller	
13. Safety Amps, Radiation Monitors, Servo Unit	
14. Annunciator	
15. Safety Amps, Magnet Power	
16. UPS	
17. Recorders and Indicators	
18. UPS	
19. Pneumatic Tube	
20. Evacuation Alarm, T.V., Intercom System, NW12 Gamma Monitor	

Note: Emergency power is available to all of Panel No. 1A and circuits 13-20 of Panel No. 1.

8.1.4 Lightning Protection

Lightning protection for the facility is described in Section 13.2.8.1 of this report.

8.2 Emergency Electrical Power Distribution

8.2.1 Design Basis

The design basis of the emergency electrical distribution system is to provide power to selected loads for a minimum of one hour following loss of off-site power. The selected loads are those necessary for lighting, communications, monitoring the reactor in a shutdown condition, and decay heat removal. The choice of a minimum of one hour is based on providing reactor information to the operator for a sufficient period following the power loss. This assures that the reactor is shut down as designed and the core is receiving adequate cooling.

8.2.2 System Description

The emergency electrical power distribution system consists of a 5 kVA (208-120 volt/three-phase) motor-generator set, the generator starting controls, a time-delay switching mechanism, transfer switches, circuit breakers, and a bank of sixty lead-calcium storage cells. The batteries are rated for 577 ampere-hours at an eight-hour discharge rate. With a nominal battery load of 72 amps, the battery bank has sufficient capacity to provide selected instrument and pump power for approximately eight hours following the loss of both external electrical power feeders. If the total battery load is increased to 150 amps, the battery bank will last for approximately three hours. A minimum of one hour of emergency power is the design requirement for this system.

The 130 volt d.c. storage battery power supply is connected to the emergency power system through a 2-pole, 200-amp fused disconnect switch. The power supply then connects to a 2-pole, 100-amp switch at the motor-generator set and to a 2-pole, 100-amp circuit breaker at the emergency lighting panel. The system is normally aligned so that power will be available to start the MG set, if needed. A separate line from the batteries supplies d.c. voltage to operate the d.c. motor-drives that operate each of the two main 13.8 kV circuit breakers. The batteries also supply d.c. lights in the utility room.

8.2.3 System Operation

The normal alignment of the emergency power system is as follows: 1) Panel No. 2 supplies 120/208 volt power to the emergency lighting panel transfer switch and thence to the emergency lighting panel itself, and 2) the 45 kVA transformer provides 120/208 volt power to both the motor-generator starter and to the Panel No. 1/1A transfer switch (and thence to Panel No. 1, Panel No. 1A, and pump MM-2). The MG starter is fed through a 30-ampere fused switch. The switches in Figure 8-1 are depicted in this configuration. When normal power fails, the following automatically occurs:

a) The emergency lighting panel transfer switch immediately shifts the emergency lighting panel from Panel No. 2 to the batteries.

b) The motor-generator starts after a twelve second delay. The delay is to prevent the MG set from starting during short duration power outages.

c) The Panel No. 1/1A transfer switch shifts to the MG set output. Hence, Panel No. 1 (Circuits 13-20), Panel No. 1A, and auxiliary coolant pump MM-2 are supplied with emergency power. The latter can be used for decay heat removal.

When normal power is restored, all transfer switches return to their normal positions and the relay at the motor-generator set is energized thereby stopping the unit.

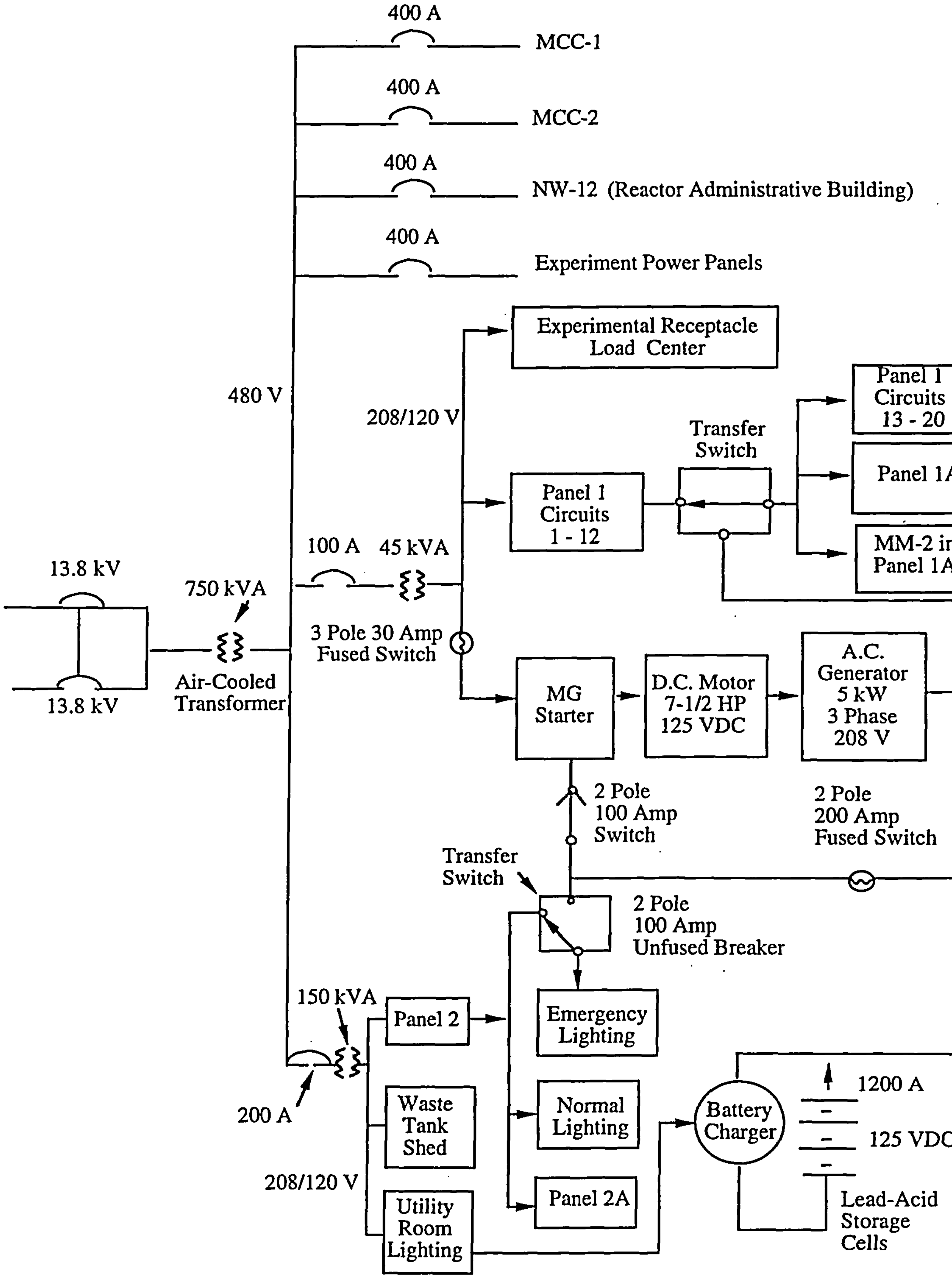
400 A
MCC-1
400 A
MCC-2
400 A
NW-12 (Reactor Administrative Building)
400 A
Experiment Power Panels
Experimental Receptacle Load Center
480 V
208/120 V
Transfer Switch
Panel 1 Circuits 13 - 20
Panel 1A
Panel 1 Circuits 1 - 12
MM-2 in Panel 1A
100 A
45 kVA
13.8 kV
750 kVA
3 Pole 30 Amp Fused Switch
Air-Cooled Transformer
13.8 kV
A.C. Generator 5 kW 3 Phase 208 V
MG Starter
D.C. Motor 7-1/2 HP 125 VDC
2 Pole 100 Amp Switch
2 Pole 200 Amp Fused Switch
Transfer Switch
2 Pole 100 Amp Unfused Breaker
150 kVA
Panel 2
Emergency Lighting
Waste Tank Shed
Normal Lighting
1200 A
Battery Charger
200 A
125 VDC
208/120 V
Panel 2A
Utility Room Lighting
Lead-Acid Storage Cells

Figure 8-1

Chapter 9

Auxiliary Systems

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Chapter 9

Auxiliary Systems

9.1 Ventilation, Heating, and Air Conditioning

9.1.1 Design Basis/Functional Requirements

MITR ventilation is provided by a single-pass system that supplies the entire containment building (~200,000 cubic feet) with fresh, climate-controlled air. The minimum stack exhaust flow is 7500 cfm. This includes dilution air that is introduced to the stack. The flow through the building is at least 3750 cfm. This provides a mean time for building air turnover of about 50 minutes and a half-life of ventilation of about 38 minutes. Functions fulfilled by this system include:

a) Maintaining containment building pressure slightly below atmospheric so that any leakage is inward.

b) Directing any airborne radioactivity away from personnel and into filtered exhaust ducts.

c) Permitting rapid sealing of the containment building upon detection of any abnormal radioactivity.

d) Providing both fresh air circulation and temperature and humidity control.

9.1.2 System Description

9.1.2.1 Inlet Air System

A schematic diagram of the heating, ventilation, and main air-conditioning system is shown in Figure 9-1. Outside air, which is drawn into the building at the rate of approximately 4000 cfm, enters the ventilation duct through fixed louvers and a coarse wire screen in the back wall of the utility room adjacent to the containment building. The air then passes through a set of remote-operated louvers, whose control is tied directly to the intake fan control circuit so that the louvers open and close whenever the fan starts and stops.

The air flow then enters a six-foot square duct that contains filtering, preheating, cooling, and reheating elements. The filter bank consists of several filter units arranged with an inclined manometer that measures the air flow pressure drop across the entire bank. The purpose of the filters is to prevent dust and other airborne particulates from entering the containment building.

The purpose of the preheat coils is to raise the temperature of the incoming air above the freezing point so that any steam that may have condensed in the reheat coils will not freeze and cause damage. A freeze-up control monitor measures the air temperature after it leaves the preheat steam coils. If the temperature of the pre-heated air drops below 38°F, a control room annunciator alarm sounds. If the air temperature continues to drop, the freeze-up control unit will secure the intake fan at approximately 35°F.

The cooling elements, besides being used to cool the incoming air during the summer, are used to remove moisture and to provide humidity control. There are two sets of cooling coils, mounted in banks located one above the other. Both are downstream of the freeze-up control monitor. Each set, which consists of a pair of coils, is connected to one of two twenty-ton air conditioning units that are interlocked with the ventilation system's intake fan control circuit so that they operate only when the fan is running. The two twenty-ton air conditioning units are air-cooled and do not utilize the MITR's secondary coolant system as a heat sink.

The reheating elements are the major source of heat. Temperature is controlled by a thermostat mounted in the containment building. The system automatically employs either the heating or the cooling elements to maintain temperature. Both the preheat and reheat coils are supplied with steam that is generated by the MIT Department of Facilities.

After passing through the filtering system and over the heating and cooling elements, the air is drawn through a 4000 cfm centrifugal intake fan. The air then passes through a fast-closing butterfly damper and an auxiliary damper into ducts which discharge into the reactor building. The fast-closing butterfly damper is operated by a hydraulic system while the auxiliary damper is opened manually and closes by the force of gravity. Both dampers remain open during normal operation of the ventilation system. If necessary, each can be closed rapidly by pushbuttons on a panel in the control room. A lanyard located inside the containment near the auxiliary damper may also be used to hand-trip it. In addition, if the main intake damper does not close within a specified time interval following receipt of a trip signal, the auxiliary damper will close automatically.

A three-position switch (open, weekend-open, and close) that is mounted in the control room provides remote control of the main intake damper. The switch is in the open position during reactor operation. Plenum gas and particulate monitors, as discussed in Section 9.1.5 of this report, will cause the main damper to close in the event that abnormal radiation levels are detected in the air that is about to leave the building. Whenever the building is unattended, the switch is in the "weekend-open" position. This means that if ventilation should be lost, the intake damper will close thereby ensuring that no backflow could be caused by temperature changes within the building. In addition to the "open" and "weekend-open" positions, the switch has a "close" position that allows the operator to close the damper manually from the control room.

9.1.2.2 Internal Distribution System

The internal air distribution system is designed in accordance with good health physics practice. Air is supplied to areas where no work involving radioactivity is performed, drawn through areas where work is done, and then exhausted to filtered ducts. In this way, potential exposures to personnel from airborne sources are minimized. The distribution system is as follows:

a) Air is supplied to the main reactor floor by circumferential ducts that run under the catwalk that encircles the interior of the containment building.

b) Air is exhausted from the main floor of the reactor building by drawing it into registers at the base of the reactor face. The exhaust air passes through a common duct and then enters the plenum chamber that is located in the equipment room on the basement level.

c) Air is supplied to the reactor basement floor from vertical ducts that are in turn supplied from the circumferential ducts on the main reactor floor. One vertical duct is on the east side of the reactor room. It discharges into the basement set up area. The other is on the west side and it discharges into the equipment room.

d) Air is exhausted from the basement setup area and equipment room through exhaust registers located in these areas.

e) The fission converter medical therapy room on the main floor of the reactor building is supplied with fresh air from the main floor. Air exhausts from this room through a duct into the exhaust system.

f) The medical therapy room in the basement is normally supplied with fresh air from the basement discharge vent. In addition, there is a booster fan that may be used to augment the supply of air. Exhaust flow is provided by one of the auxiliary fans that are described in Section 9.1.2.3 of this report.

g) Air circulation within the upper level of the containment can be promoted by a fan that is located at the top of the reactor building dome. However, this fan is rarely needed, and its principal use is to prevent stratification during building pressure tests.

h) Air circulation to the control room is provided by a booster fan that is interlocked with the main ventilation fans. It forces fresh air from the inlet duct on the main floor into the control room and then out through a vent in the door leading to the equipment room. This vent has a cover plate that can be used to seal the opening. The ventilation for the control room is therefore reasonably isolated from that of the rest of the building, and the control room is semi-gastight with the booster fan off and the exhaust vent in the door sealed.

i) Air circulation in the spent fuel storage room, which is located in the reactor basement area, can be provided by a wall-mounted fan that draws air through the door of the room and exhausts it to the equipment room where it is drawn into the exhaust plenum. However, this fan is not normally needed.

9.1.2.3 Auxiliary Fans

Auxiliary fans exhaust air from the primary chemistry room, the basement medical therapy room, the reactor floor hot cell, and the secondary chemistry area. Exhaust ducts from the areas served by these fans join the main exhaust plenum in the equipment room. There are also two "Flexhaust" hoses that connect to the suction of the secondary chemistry area's fan. There is a four-inch hose that extends from the reactor top down to the equipment room and a six-inch hose located in the equipment room. Both hoses are used to provide local ventilation for maintenance activities.

The auxiliary fans are normally operated by a single push-button on a panel in the control room, but individual switches located near the fan units can be used to isolate any of the fans from the single start-stop control. Also, the Flexhaust hoses are equipped with manually-operated dampers so that they can be shut off. All auxiliary fans are interlocked with the building ventilation and will trip if the exhaust fan stops.

9.1.2.4 Core Purge Blower

The core purge blower is located in the equipment room. It takes a suction from the air volume right above the core tank pool, through the off-gas pipe, through the top of the primary storage tank, and into the core purge filter jars. The blower removes Ar-41 and fission product gases such as xenon and particulates such as cesium. It is interlocked with the building ventilation and will trip when the main or auxiliary dampers close or the exhaust fan stops. Detection and isolation functions of the core purge system are described in Section 9.1.5.2 of this report.

9.1.2.5 Pneumatic Blower

The pneumatic blower is used to operate the pneumatic tube irradiation facilities. It applies a vacuum to the inner or outer end of a sample tube by means of a bank of solenoid valves. This vacuum either draws the polyethylene sample cylinder (or other approved container) up through the pneumatic tubes into the graphite reflector region, or pulls the cylinder out of the tubes to either the primary or secondary chemistry areas. It is also interlocked with the building ventilation to trip whenever the main or auxiliary dampers close or the exhaust fan stops. This trip can be manually overridden in order to eject heat-generating or radiation-sensitive samples.

9.1.2.6 Exhaust Air System

The exhaust air, which includes the discharge of the auxiliary fans as well as that of the pneumatic tube blower and the core purge blower, enters the plenum upstream of a redundant effluent monitoring system. Gaseous and particulate activities in the plenum are monitored by continuous air sampling. Radiation levels are indicated in the control room, and if operating limits are exceeded, the plenum monitors will trip causing the intake and exhaust butterfly dampers to close, and the intake, exhaust, and auxiliary fans to stop.

Immediately beyond the plenum monitors is a long hold-up chamber with a perforated baffle at its entrance that is designed to provide a time delay of several seconds before air passing the monitors can reach the butterfly damper in the exhaust duct. This delay is sufficient to ensure that, in the event of a plenum monitor alarm and a ventilation trip, the butterfly damper will close and seal the building before the detected activity can be released to the environment. Once through the hold-up chambers, the exhaust air passes the auxiliary exhaust damper and the main, fast-closing, butterfly exhaust damper. It then passes through the ΔP control damper, the exhaust filters, the exhaust fan, and the stack isolation damper, and flows out the stack.

The fast-closing butterfly exhaust damper is, like its counterpart in the intake ventilation duct, hydraulically operated. It has an "open" and "close" remote-control switch that is located in the control room.

The gravity-operated auxiliary exhaust damper is located upstream of the butterfly damper and is held open by a solenoid mechanism. If voltage to the solenoid is lost, the damper is released and falls by the force of gravity against a rubber gasket. The solenoid mechanism may be tripped by a pushbutton in the control room, a switch in the equipment room near the exhaust plenum, or an interlock with the main exhaust damper. The interlock ensures that if the main damper does not close within a specified time interval following receipt of a trip signal, the auxiliary damper will close. Once closed, the auxiliary damper must be opened manually.

An exhaust control damper is located at the stack base upstream of the filtering units. It is used to throttle the overall flow of air through the ventilation system. The position of this damper is adjusted to maintain the building differential pressure by balancing the exhaust air flow with that of the intake. The damper's position can be adjusted remotely from the MITR control room or by means of a geared hand wheel that is located in the stack base.

The exhaust filtering unit consists of both a bank of coarse filters and one of absolute filters. The function of the coarse filters is to remove the larger particles from the exhaust air to prevent fouling of the absolute filters. The purpose of the absolute filters is to remove particulates. The pressure drop across each filter bank and the total drop across the entire unit are monitored by three manometers. This information is used to determine the condition of the filters.

Once past the filter unit, the discharged air proceeds through the stack exhaust fan and up through the 150 foot stack from which it is dissipated to the atmosphere. The stack exhaust fan is rated at 8000 cfm. A flow of approximately 4000 cfm of air passes through the reactor building during normal operation. Two eight-inch inlet bleeder ducts located immediately before the exhaust fan allow outside air to mix with the discharged air. This mixing-air dilutes the discharged air and lowers the radioactive concentration of any effluent gas.

External stack effluent monitors are located downstream of the exhaust fan. These detectors monitor gas and particulate matter that may have passed the filter banks. The air then passes upward through the 34-inch diameter opening in the bottom of the stack and past the pressure relief system's stack isolation damper. An area radiation monitor mounted on a platform 40 feet above the stack base monitors the air for gross radiation level before it reaches the atmosphere. All effluent instruments read out in the control room.

9.1.2.7 Control Room Air Conditioning

The MITR control room is equipped with an air conditioning unit that provides instrument cooling by circulating cool air through the electronic cabinets, around the room, and back through the air conditioner.

9.1.3 Operational Analysis and Safety Function

Interlocks associated with the ventilation system are listed in Table 9-1 of this report. Those that result in isolation of the containment building are important to safety because they block an effluent path. See Section 7.7.2 of this report.

9.1.4 Containment Differential Pressure Systems

Containment differential pressure is defined as the difference between the pressure inside the reactor building and atmospheric pressure. Two independent systems are used to measure this parameter. The first system reads out in the control room and is used to provide appropriate alarms, scram, or ventilation fan shut-off in the event of pressures that are outside the normal operating range of -0.18 to -0.34 inches of water. The second system, which reads out in the instrument cabinet in the utility room, provides a remote indication of building overpressure conditions and enables one to obtain this information without having to enter the containment building.

The first system consists of a differential pressure meter, a ΔP recorder, and four differential pressure switches. The meter indicates containment differential pressure over a range of +0.50 to -0.50 inches of water. The sensors for this instrument are located in the utility room and the control room. The recorder monitors containment differential pressure on a scale ranging from +0.80 to -3.20 inches of water. In addition, it records the barometric pressure over a range of 28.50 to 31.00 inches of mercury. The pressure taps for this instrument are located in the waste tank shed and in the primary chemistry set-up area. Neither instrument provides any alarm functions.

All alarm and interlock functions are initiated by the four differential pressure switches. The first, XPS-1, provides an abnormal ΔP annunciator alarm if the differential pressure falls below -0.10 inches of water. In addition, this switch provides an interlock in the withdraw-permit circuit that must be cleared before a "reactor start" may be obtained. The second switch, XPS-2,

Table 9-1

Ventilation System Interlocks

Interlock	Required Conditions to Activate or Clear
Radiation Monitors / Dampers Interlock	High radiation level trip on any plenum gas or particulate monitor or a plenum monitor blower off closes main dampers. (Fans shut off on damper/fan interlock below.)
Core Purge Radiation Monitor / Air Purge Secured	High radiation level trip on core purge monitor stops purge blower, which closes core space isolation valves.
Main Fan Sequence Interlock	Intake fan must be operating before exhaust fan will start.
Auxiliary Fans / Exhaust Fan Interlock	Auxiliary fans stop when exhaust fan stops.
Core Purge Blower / Exhaust Fan Interlock	Core purge blower stops when exhaust fan stops.
Pneumatic Blower / Exhaust Fan Interlock	Pneumatic blower stops when exhaust fan stops (bypass provided).
Control Room Intake Fan Interlock	Fan stops when intake fan stops.
Freeze-Up Control	Intake fan stops when temperature of air leaving preheat coils drops to approximately 35°F.
Building ΔP Interlock	Intake fan stops if reactor building pressure exceeds -0.4" H_2O or +1.2" H_2O relative to atmospheric.
Intake Damper Low Oil Pressure Interlock	Main intake damper closes when oil pressure decreases to approximately 700 psig.
Exhaust Damper Low Oil Pressure Interlock	Main exhaust damper closes when oil pressure decreases to approximately 700 psig.
Intake Louvers Interlock	Intake louvers open when intake fan is started.
Main Damper / Auxiliary Damper Interlock	If a main damper fails to close on a radiation monitor trip signal, the associated auxiliary damper will close after a preset time delay.
Damper / Fan Interlock	Stops intake, exhaust, and auxiliary fans if either main damper or auxiliary damper or the pressure relief damper closes.
Intake Fan / Main Intake Damper Interlock	The main intake damper closes if the intake fan stops during a shutdown condition. (Intake damper control switch must have been in the weekend-open position.)
Reactor Floor Hot Cell / Fire Detection Interlock	The reactor floor hot cell blower stops if either hot cell fire detection unit alarms.
Exhaust Damper / Helium Supply	GV-53 isolates helium manifold when exhaust damper shuts.
Exhaust Damper / CO_2 Supply	OV-9 isolates CO_2 manifold when exhaust damper shuts.
Intake Fan /AC	Twenty-ton A/C units operate only if intake fan is running.

trips the ventilation fans if the differential pressure exceeds -0.40 inches of water. The third switch, XPS-3, which measures the pressure difference between the waste tank shed and the control room, provides both a building overpressure alarm and an automatic scram at an overpressure of no more than 3.00 inches of water. The fourth switch, XPS-4, senses the same pressure difference but trips the ventilation fans at an overpressure of 1.20 inches of water. All pressure lines in this ΔP system are 1/4-inch diameter tubing.

The second differential pressure system is located in the utility room remote instrument cabinet. It consists of two Magnehelic pressure gauges. One gauge, which is considered low-range, measures pressures from 0.0 to 8.0 inches of water while the second, which is considered high-range, monitors pressures from 0.0 to 3.0 psig. Containment pressure is sensed through a building penetration located above the instrument cabinet. In the unlikely event that an accident precludes entry to the reactor building, this system could be used to obtain indication of the building pressure and to determine if use of the containment building pressure relief system is warranted. In addition, the system can provide a backup pressure indication during building pressure tests.

All differential pressure system containment penetrations contain 1/32-inch diameter orifices. Were a pressure sensor line to rupture, these orifices would restrict flow through the broken line so that the building leak-rate specifications would still be met.

The operation of the differential pressure system is checked according to the following schedule. The abnormal ΔP alarm and interlock are tested as part of the startup checklist, which is done at least quarterly. The building overpressure alarm and scram are tested at least once each year.

9.1.5 Radiation Monitors

9.1.5.1 Effluent Air Monitors

Redundant gaseous and particulate monitors are installed both upstream and downstream of the exhaust plenum. These are referred to as the "plenum" and "stack" radiation monitors respectively. (Note: The latter should not be confused with the stack area monitor, which is located forty feet up the ventilation exhaust stack and has separate circuitry.) Each monitor uses a pancake-type GM detector within a lead shield to reduce background. In addition, the particulate monitors contain filter assemblies with both HV-70 filter paper and activated charcoal filters.

Each set of particulate and gaseous monitors has its own blower and instrumentation. If flow is lost through a monitor set, an alarm will activate. In addition, if flow has been lost through a plenum set, the ventilation system will be automatically isolated. A high level trip on any one of these four plenum monitors will, besides energizing a high radiation alarm on the annunciator panel, cause the building ventilation system's intake and exhaust fans to stop and the isolation dampers to close before the sampled gas can pass by the exhaust damper. This ensures that whatever the abnormal activity detected, it will be trapped within the reactor building. Depressing a major scram pushbutton also produces the response of stopping and isolating the building ventilation system.

9.1.5.2 Core Purge Monitor

The core purge monitor is described in Section 5.2.1.7 of this report and shown in Figure 5-1. The monitor uses a pancake-type GM detector that monitors activity in the core purge flow through a stainless steel window. A blower located in the equipment room draws air from the reactor containment building through an inlet filter, over the water in the main core tank, and past the detector. The air is then discharged through an outlet filter to the main ventilation exhaust duct.

The point at which the core purge flow merges with the building exhaust ventilation is located upstream of the plenum effluent monitors.

A high level trip on the core purge monitor, in addition to energizing an alarm on the annunciator panel, will automatically turn the core purge blower off, thereby closing valves that isolate the core purge system. This function allows building ventilation to be maintained even if high activities are present in the air over the main core tank. However, the reverse is not possible. The core purge blower is interlocked with the building ventilation so that if ventilation is secured, the blower automatically stops.

9.1.6 Interlocks

Table 9-1 lists the interlocks associated with the ventilation system.

9.2 Reactor Fuel Storage and Handling

9.2.1 Storage of Unirradiated Fuel

Unirradiated fuel elements may be stored in any of the following locations subject to the provisions of the MITR Physical Security Plan:

a) In the reactor core provided that the reactivity is below the shutdown margin,

b) In the cadmium-lined fuel storage ring attached to the flow shroud, or

c) In the storage safe in the reactor containment building.

In addition, individual fuel plates may be stored in the storage safe. A maximum of four new fuel elements or the equivalent of two new fuel elements including loose plates (maximum of fifteen plates) may be outside of the above storage areas except during the process of receiving or shipping fuel from the site in approved containers. The value of k-effective for all storage areas except the core shall be less than 0.90.

The principal issues associated with the storage of unirradiated fuel are those of security and inadvertent criticality. The former is addressed in the MITR Physical Security Plan. Criticality safety is ensured by proper design and use of each storage location. Specifically, the reactor itself is shielded, and approved written procedures are observed to assure that it is loaded properly. The fuel elements in the cadmium-lined storage ring in the core tank are neutronically isolated from the reactor by the cadmium of each individual box and by the six shim blades that surround the reactor core. The storage safe contains a square array of storage positions with each row separated by cadmium. The limit on the total number of elements and/or loose plates that may be outside of an approved storage area is based on a safety factor of two. At least 8-1/3 MITR fuel elements with optimum spacing and total reflection are required for criticality.

9.2.2 Storage of Irradiated Fuel

Irradiated fuel may be stored in any of the following locations subject to the provisions of the MITR Physical Security Plan:

a) In the reactor core provided that the reactivity is below the shutdown margin,

b) In the cadmium-lined fuel storage ring attached to the flow shroud,

c) In the fuel element transfer flask or other proper shield within the controlled area,

d) In the fission converter, or

e) In the spent fuel storage pool.

The above locations are all within the reactor containment building. Security issues are addressed in the MITR Physical Security Plan. Criticality is not possible in the fuel element transfer flask because only one element can fit in it at a time. The k-effective of the fission converter is a function of fuel burnup and coolant type (H_2O/D_2O). However, for all cases, its k-effective is well below 0.90. The spent fuel pool contains storage racks which are arrays of open-ended boxes, each box lined with cadmium. This design assures the value of k-effective is less than 0.90 for the spent fuel pool.

Shielding for each of the above storage locations (except the transfer flask) is provided by water. The transfer flask is a lead-filled annulus with a lead thickness of 10.9 inches. This is sufficient to reduce the dose from an irradiated element by a factor of at least 10^6.

9.2.2.1 Spent Fuel Storage Pool

The spent fuel storage pool is a 21-foot deep steel-lined concrete tank in the reactor building's basement. It is used to store spent fuel elements, used control blades and other irradiated core components that are no longer in use. The steel liner is a 3/4-inch thick hot-rolled mild steel cylinder with an inner diameter of eight feet. It extends 27 inches above the level of the containment's basement floor. The bottom of the tank is a two-inch thick slab made of the same type of steel. All exposed surfaces of the steel tank and rings have been coated with an epoxy paint to prevent rusting.

Spent fuel may be stored at the bottom of the pool in storage racks. Three racks, each an array of open-ended boxes in an aluminum frame, are currently installed. The boxes, which are 3.75 inches square on the inside, are constructed of an aluminum-cadmium-aluminum sandwich. The pool is filled with high purity deionized water which serves both to remove decay heat from the spent fuel and to shield personnel from the gamma radiation produced by the elements.

The purity of the pool water is maintained by a cleanup system that is constructed entirely of non-ferrous materials. Water analyses, including pH, chlorides and conductivity, are performed at least quarterly. The beta-gamma and tritium activity of the water is also measured at least quarterly. The filter system is equipped with two interlocks that cause the pump to trip on receipt of either a low pool level or a leak alarm. The purpose of these interlocks is to prevent flooding of the basement in the event of a pipe break in the system.

The spent fuel pool storage room is equipped with an area radiation monitor that alarms both locally and in the control room in the event of an excessive radiation level.

9.2.3 Handling of Fuel Elements

All proposed changes to an existing configuration of fuel elements in the core tank or fission converter fueled region must be reviewed and approved by both a senior reactor operator and the reactor superintendent. This review, which is documented, includes a determination that the applicable technical specifications will be met after the proposed change, specification of the movement sequence, and delineation of any measurements (such as verification of reactivity changes) that are to be made either during or after the change. For the MITR core, the relevant technical specifications are those that relate to the core power distribution, limiting operating conditions, and shutdown margin. For the fission converter, they would be those that pertain to limiting operating conditions for the fueled region. For fuel transfers to the spent fuel storage pool, they would be the ones that concern power history and hence decay heat generation.

The movement of fuel elements is governed by written procedures. Only one element may be moved at a time. Moreover, movements are performed using specialized tools. These include:

a) Lifting Hook: This tool is used to transfer elements in and out of approved locations within the approved configuration. A hook with a spring-loaded latch attaches to the bale of the element. The latch is closed prior to movement so that the element cannot fall off the hook. The spring ensures that the latch cannot inadvertently open during movement.

b) Inversion Tool: MITR fuel elements can be rotated axially to improve burnup. This operation requires that elements be flipped. A special tool that protects the element during the inversion process is used.

c) Transfer Basket: The transfer of fuel elements from the core to the spent fuel storage pool requires that the element be raised from the core through a penetration in the reactor top lid. Elements are protected by use of a transfer basket during this operation.

9.2.3.1 Transfer of Spent Fuel

The MITR core tank is not connected by a canal to the spent fuel storage pool. Hence, the transfer of fuel elements from the core and the fission converter is done by use of a shielded transfer flask. This in turn means that the power history of the fuel element to be transferred must be such that the dose rate on the flask surface and the decay heat generation rate of the element are acceptable.

9.2.3.2 Preparations for Shipping

The procedures and equipment described above for the handling of MITR fuel are employed during the loading of spent fuel shipping casks. In addition, the separate procedures that are a part of the available cask's license are followed as are the regulations of 10 CFR 71, 10 CFR 73, and the U.S. Department of Transportation.

9.2.4 Fuel Self-Protection

The length of time that an element remains self-protecting depends on its power history. MITR elements that have been in-core for a normal cycle (3 - 5 years) will remain self-protecting for at least 10 years. This has been confirmed by measurement of selected elements in air [9-1].

9.3 Fire Protection

9.3.1 Passive Components

The MITR containment building is made of steel and concrete. Most of the structures contained within it are made of aluminum or other fire-resistant materials. Also, the large volume of water in the core tank would protect the core from a fire. Finally, the reactor safety system is fail-safe. If it were to be damaged by fire, the control devices would be dropped into the core by gravity.

9.3.2 Active Components

The MITR is equipped with a variety of fire detection devices. The equipment includes:

a) Pull Boxes: Manually-operated pull boxes are located at the reactor floor, reactor basement, and equipment room. These sound in the control room and trip a distinct alarm that is audible throughout the building.

b) Area Detectors: Smoke detectors are located in the control room, equipment room, and exhaust plenum. These also sound in the control room and trip a distinct alarm that is audible throughout the building. If the building is unattended, the alarm is forwarded to the MIT Campus Police.

c) Local Detectors: The reactor floor hot cells are equipped with heat sensors which trip the hot cell blower, and alarm as do the smoke detectors described above.

All fire detection devices (pull boxes, smoke detectors, and heat sensors) are monitored from a dedicated panel that is located in the reactor control room. Panel features include a trouble monitoring circuit that alarms if a detection device is malfunctioning (open or short-circuit, loss of power, or undervoltage), a switch to perform alarm tests as well as to silence both the fire and trouble alarms, and a reset switch. If any switch is not in its normal operating position, a trouble lamp and alarm are energized.

In addition to these dedicated fire detection devices, there is a closed-circuit television system that allows the control room operator to monitor the experimental areas on the reactor floor. Also, the control room itself can be monitored by closed-circuit television from outside the containment building.

9.3.3 Prevention Components

Inventories of flammable materials such as solvents, wood, and paper are minimized. If an experiment requires use of a flammable or explosive substance, its use and allowed quantity are carefully reviewed. For example, such an issue is addressed in the safety review contained in the Appendix to Chapter 13 of this report. In addition, the building is periodically inspected for industrial safety issues.

The MITR has written procedures that address the actions to be taken in the event of a fire. If the fire is identified in its incipient stage, CO_2 extinguishers could be used to suppress it.

These are located throughout the building and are inspected and serviced regularly. For fires that are beyond the incipient stage, assistance would be requested from the local fire department. Personnel who would respond to such a fire tour the facility on a regular basis. Release of radioactive material during a fire is not expected because the ventilation ducts will seal if abnormal levels of radioactivity are present in the effluent air. Moreover, the system is fail-safe in that if the detectors that monitor the effluent were damaged by a fire, the ducts would be sealed. The same is true if the damper control mechanisms themselves were damaged.

9.4 Communication Systems

The principal means of communication is via telephone. Units, most with separate numbers, are located in the reactor control room, on the main reactor floor, on the reactor top, and in both medical therapy areas. These phones are all part of the MIT telephone system. Calls can be both made and received on these phones from locations within and outside MIT. The MIT phone system is supplied by emergency power (diesel generator) so that there is no disruption of service even if there should be a loss of the normal power supply. Other provisions for communication include an intercom system with a master unit installed in the control room and with many slave units located throughout the reactor facility. A call can be initiated at either the master unit or at a slave station. No communications are possible between slave units. Reactor emergency power supplies the intercom system in the event of loss of the normal electric power supply.

9.5 Byproduct and Special Nuclear Material

9.5.1 Byproduct Material

MIT maintains a separate license for the possession of byproduct material. That license is administered by the MIT Radiation Protection Office and is not a responsibility of the MITR

Operations Staff or the MIT Reactor Radiation Protection Office. Most uses of byproduct material at MIT are performed under this license. The MITR operating license also allows the receipt and use of byproduct material. The primary purpose of this provision is to facilitate the receipt of radioactive materials for materials analysis. The limitations on this provision are:

a) Material must be atomic number Z = 3 to 83 and in solid form only,

b) The total inventory of materials from non-reactor sources may not exceed 100,000 curies,

c) The inventory for any one sample or specimen is limited to 1000 curies, and

d) Each sample or specimen is limited to radiation flux (unshielded) of 100 rads per hour, at one meter from the sample or specimen.

Material received under the above provision of the MITR operating license is normally stored in the reactor floor hot cell. However, other approved storage locations may be used depending on the amount of activity involved. Written procedures govern the receipt, storage, and inventory of this type of material.

9.5.2 Special Nuclear Material

MIT also maintains a special nuclear material (SNM) license. Material held under this license may be administered by either the MIT or the MITR Radiation Protection Office depending on the use of the material. Nearly all of the SNM held at the reactor site is in safe storage pending return to the U.S. Department of Energy.

The MITR Operating License does authorize the possession of some SNM. This is limited to the U-235 utilized in conjunction with operation of the reactor and to [redacted] plutonium-beryllium (Pu-Be) sources. The latter are used for reactor startups if the photoneutron source has decayed away. The fuel is stored as described in Section 9.2 of this report. The Pu-Be sources are stored in shielded containers inside a locked enclosure that is within the reactor's restricted area.

9.5.2.1 Authorization for the Possession of U-235

The major uncertainty in the above estimate is the inventory of material awaiting return to the U.S. Department of Energy (DOE). Historically, this has varied because of factors beyond MIT's direct control. All of the material listed above, except the four fresh elements and materials in item (e), is maintained self-protecting.

9.6 Cover Gas Control and Processing

9.6.1 Primary Coolant System Core Purge

This system is described in Sections 5.2.1.6, 5.2.1.7, 9.1.2.4, and 9.1.5.2 of this report.

9.6.2 Heavy-Water Reflector Helium Cover Gas

This system is described in Section 5.3.6 of this report.

9.6.3 Graphite Reflector CO_2 Cover Gas

The graphite reflector is described in Section 4.2.3.3 of this report. It is blanketed by CO_2 or a suitable inert gas to prevent air from entering the graphite region.

The CO_2 system supplies a constant pressure gasholder in the equipment room which in turn maintains a CO_2 blanket/purge on the entire graphite reflector region. The volume in the gasholder is replenished automatically by the CO_2 system's manifold. Pressure switches on the discharge side of the gasholder actuate annunciator alarms to warn the operator of high or low pressure conditions in the system. The gasholder has the same overpressure protection as the D_2O reflector's helium gasholder.

9.7 Auxiliary Coolant Systems

The MITR's heavy-water reflector and shield coolant system are discussed in Sections 5.3 and 5.4 of this report. The latter may be used to provide cooling to experimental facilities.

9.8 Radioactive Material Storage

9.8.1 Control and Storage of Radioactive Waste

See Section 11.2 of this report.

9.8.2 Control and Storage of Reusable Radioactive Components

The reuse of radioactive components is an important means of reducing the volume of radioactive waste. Tools and other reusable items are decontaminated, or if short-lived radionuclides are involved, stored temporarily. Additional information is given in Section 11 of this report.

9.9 Control of Effluents from Experimental Facilities

The operation of experimental facilities may produce solid, liquid, or gaseous effluents. Experiments are designed as described in Section 10.3 of this report to minimize all effluents. The principal ones of concern result from the exposure of air to both neutron and gamma fluxes. Accordingly, a CO_2 system is used to minimize both Ar-41 and nitrous oxide formulation. The CO_2 system, in addition to supplying the graphite reflector region also supplies many of the experimental facilities. The CO_2 displaces air and thus greatly reduces the formation of argon-41, nitrous oxide, and condensed moisture.

A manifold located outside the containment building supplies CO_2 via an isolation solenoid valve in the utility room. The CO_2 is distributed to a variety of experimental facilities through regulated flow meters.

CO_2 flow is also supplied to the pneumatic tubes in the reactor basement and to the graphite gasholder in the equipment room. (Note: Helium or other appropriate gas could be used in lieu of CO_2.)

9.10 Compressed Air System

Figure 9-2 is a schematic diagram of the compressed air system. The purpose of the system is to supply compressed air for loads in the reactor containment building (including instrument air) and in two laboratories that are located in Building NW12 (the administration building for the MIT Nuclear Reactor Laboratory). There are two compressors, CM-1 and CM-2. Both are located outside the containment building. Compressor CM-1 supplies the laboratories and serves as a backup for CM-2. Compressor CM-2 supplies all loads within the containment building and the adjoining restricted area. It cannot be used as a backup to CM-1. Table 9-2 lists the components in the reactor containment building that are supplied with compressed air.

Compressor CM-2 supplies air to a 120-gallon air receiver at 120-170 psig, controlled by a pressure switch mounted on the air receiver. A relief valve will open at approximately 200 psig, should CM-2 malfunction. The air from the receiver is supplied to the containment building

Table 9-2

<u>Compressed Air System Loads</u>

1. Main personnel lock door gaskets
2. Basement personnel lock door gaskets
3. Truck lock girth seal
4. Medical facility lead and boral shutter actuators
5. 1PH1 - 2PH1 pneumatic system valves
6 Building NW13 pneumatic tube automatic isolation valve
7. Silicon irradiation facility
8. Basement medical facility water shutter operating valve
9. Basement medical facility water shutter cooling isolation valve
10. Reactor core purge isolation valves
11. D_2O blister tank drain valves
12. Reactor building pressure test metering system
13. D_2O reflector dump valve
14. Hose connections for pneumatic tools and other uses
15. Instrument air system
16. Primary storage tank fill valve from makeup water system
17. Pneumatic valves for water shutter in fission converter medical facility

and the adjoining restricted area. A compressed air low pressure alarm warns the control room operator of any underpressure, while relief valves protect the system from overpressure if a regulator should fail. The air supply line from the utility room can be closed from either side of the basement containment wall by means of manual isolation valves.

The air leaving the CM-2 receiver passes through a moisture filter and a refrigerated air dryer which removes the major portion of the moisture contained in the air. In addition, an automatic drain valve periodically cycles to drain moisture which accumulates in the air receiver. Compressor CM-1 provides air to NW12 and would provide a backup supply of air to the reactor system through a check valve arrangement if compressor CM-2 were to fail or needed to be shut down for servicing.

9.10.1 Personnel Locks

A portion of the air is regulated to ~35 psig to supply the building personnel locks and the truck lock girth seal. The personnel lock gaskets are protected from overpressure by a relief valve set at ~60 psig. If the air supply for the pneumatic inflatable gaskets of the main personnel lock fails, or should a serious gasket leak occur, CPS-1 will cause a low pressure alarm to annunciate in the control room. If neither air compressor is available, as for example during a power failure, a high pressure air cylinder that is kept in the utility room may be used as an emergency supply. A check valve permits this emergency air to be supplied only to the personnel locks and truck lock girth seal.

If both gaskets on either the main or the basement personnel lock should be deflated simultaneously, a corresponding alarm will annunciate in the control room. Inasmuch as this condition represents a loss of the containment building's seal, an automatic reactor scram will result.

9.10.2 Fission Converter Medical Irradiation Room

Pneumatic control valves for the water shutter in the fission-converter medical facility are operated by the compressed air system.

9.10.3 Basement Medical Irradiation Room

Air flow to the pneumatic cylinders that move the basement medical therapy facility's lead and boral shutters is regulated by solenoid-operated valves that are controlled from the medical control panel. If the air supply pressure is insufficient to close the shutters, a manual release lever can be pulled. This lever opens a three-way valve and vents the cylinders so that the shutters can then be drawn closed manually. A pressure switch senses the supply pressure to the operating cylinders, and if the pressure drops below its setpoint, the switch actuates a lamp on the medical control panel.

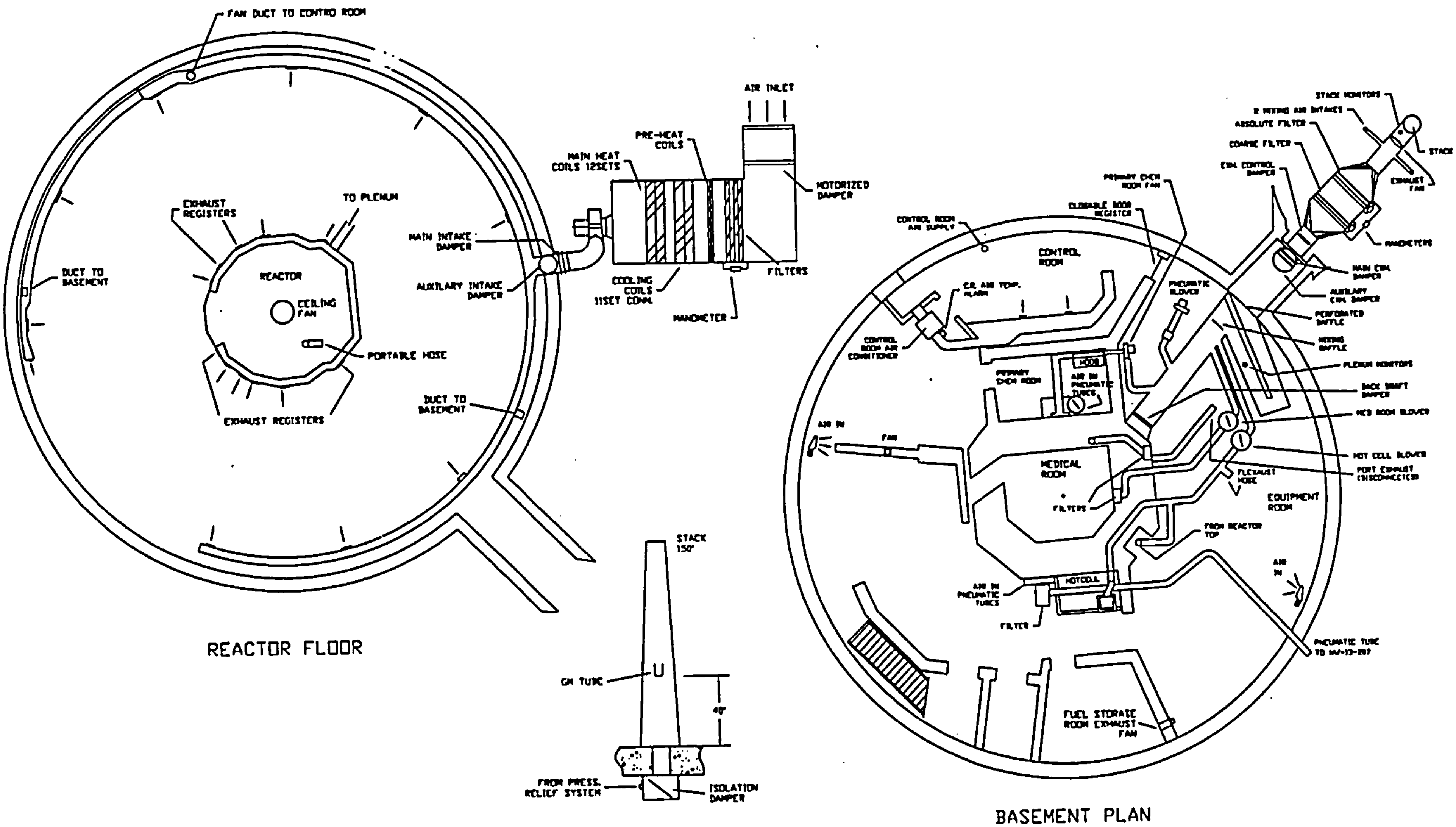
9.10.4 D_2O Reflector Dump Valve Air System

The solenoid-actuated, air-operated dump valve requires about 25 psig to remain closed. When the dump valve selector switch on the main console is switched to open or when a major scram pushbutton is depressed, a solenoid-operated valve shuts off the air supply to the dump valve and simultaneously vents the air chamber above the diaphragm. The dump valve then opens and dumps the upper portion of the reflector. An alternate means of accomplishing this is to use the emergency air bleed valve, which is also located on the main console. This action mechanically cuts off the air supply and allows the air pressure above the diaphragm to bleed off. The absence of adequate supply pressure will immediately cause the dump valve to drift open, thereby dumping the upper reflector.

Air to the D_2O dump valve operator is normally supplied by the main air compressor, CM-2, located in the utility room. If that compressor fails, or it is removed from service for maintenance, backup air is automatically supplied from backup compressor CM-1. The compressed air low pressure alarm would warn the control room operator in advance of this impending abnormal operating condition.

References

9-1 File Memo (MITR Fuel Self-Protection, February 1997)



HEATING, VENTILATION, AND
AIR CONDITIONING SYSTEMS
FIGURE 9-1

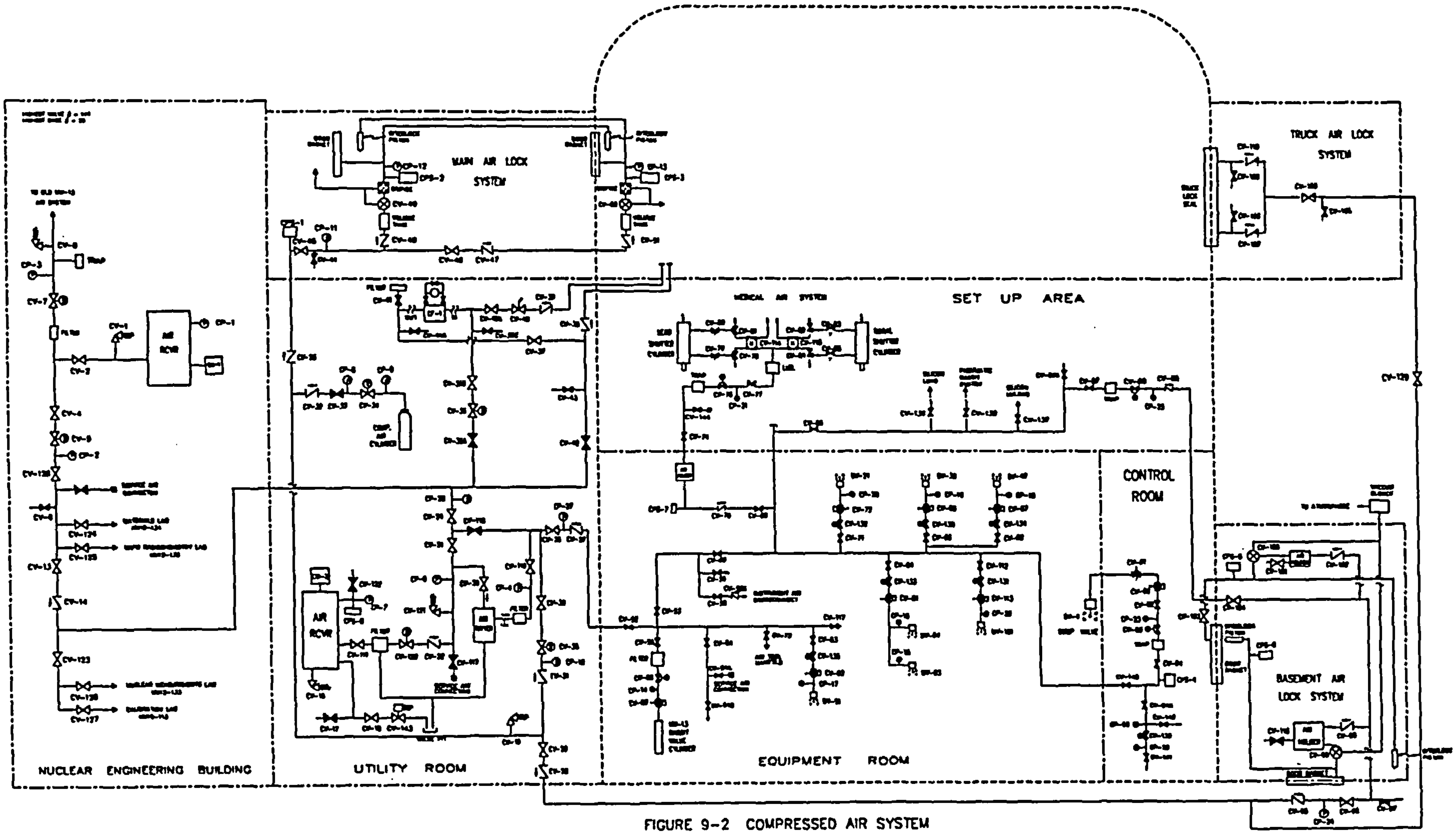


FIGURE 9-2 COMPRESSED AIR SYSTEM

Chapter 10

Experimental Facilities and Utilization

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Chapter 10

Experimental Facilities and Utilization

10.1 Summary Description

The MITR is used to support the education and research missions of the Massachusetts Institute of Technology. In addition, the facility is made available to other local-area universities and organizations as well as to hospitals. The result is a very broad research program that encompasses most aspects of neutron science and engineering. Major aspects of the program include nuclear medicine with emphasis on both neutron capture therapy and radiation synovectomy, neutron activation analysis for the identification of air pollutants and isotope ratios in geological specimens, fission engineering including digital control of spacecraft reactors, materials testing and evaluations, and teaching. The MITR is, as of this writing, one of two facilities in the United States and one of only five in the world to be engaged in patient trials for the use of neutron capture therapy to treat both glioblastoma multiforme (brain tumors) and deep-seated melanoma (skin cancer).

Experimental facilities available at the MITR include two medical irradiation rooms, beam ports, automatic transfer facilities (pneumatic tubes), in-reflector (graphite) irradiation facilities, and several in-core sample assemblies. The latter are not permanently installed. In addition, the MITR itself may be used to demonstrate techniques for the digital control of nuclear reactors. The MITR does not presently have either a cold neutron source or a thermal column. Table 10-1 lists all permanently installed facilities grouped by type. Figure 10-1 is a horizontal cross-section of the MITR in which the beam ports, automatic transfer, and in-reflector (graphite) irradiation facilities are shown. (Note: the original MIT Research Reactor, MITR-I, was built with a medical irradiation room located directly below the core. The neutron beam provided to that room was thermal. In the late 1980s, that beam was modified to be epithermal. Also installed with the original reactor was a thermal column. That facility was removed in the late 1990s and a fission

Table 10-1

MITR Experimental Facilities

A. Medical

1. Medical room beside core with horizontal epithermal beam.
2. Medical room below core with vertical thermal or epithermal beam.

B. Beam Ports

1. One 12-inch diameter port, radial, with shutter.
2. Four 6-inch diameter ports, radial, with shutters.
3. Two 6-inch diameter rotary ports, semi-radial, for sample irradiations.
4. One 6-inch diameter through-port (two port ends).
5. Six 4-inch diameter radial ports.
6. Four 4-inch diameter instrument ports.
7. Two 4-inch diameter through-ports (four port ends).

C. Automatic Transfer

1. 2PH1 - high flux pneumatic tube.
2. 1PH1, 1PH2, 1PH3, 1PH4 - intermediate flux pneumatic tubes

D. In-Reflector (Graphite) Irradiations

1. Six 3GV (graphite-vertical) irradiation facilities located in the graphite reflector.

converter was installed. The fission converter provides an epithermal neutron beam to a second medical irradiation room located on the reactor floor.)

Many types of experiments can be (and have been) conducted using the MITR. These include in more or less chronological order from the time of the reactor's initial criticality in 1958:

a) A loop for the study of organic coolants in power-generating reactors.

b) Neutron physics including refinement of neutron charge neutrality measurements.

c) Blanket neutronic studies.

d) Boron neutron capture therapy using thermal neutrons.

e) Use of neutron activation analysis for a variety of topics:

  (i) Mineral uptake in the human body using stable isotopes.

  (ii) Analysis of meteorites and lava flows.

  (iii) Identification of origin of air pollutants.

f) Studies of candidate material for the first wall of fusion machines.

g) Development and demonstration of techniques for the closed-loop digital control of spacecraft and terrestrial reactors. Techniques include signal validation, use of analytic redundancy, reactivity constraint approach, rule-based control, and period-generated control.

h) Radiation synovectomy (use of beta-emitters to treat arthritis of the knee).

i) In-core loops for the evaluation of water chemistries for both pressurized and boiling water reactors.

j) Verification of the linearity of the neutron wave equation.

k) Studies to identify both the cause and remedy for crack propagation in pressurized water reactor materials.

l) Neutron capture therapy using epithermal neutrons.

m) Radiation synovectomy using boron neutron capture therapy.

Both the monitoring and control of an experiment and its interaction (if any) with the reactor control and safety system may vary widely depending on the experiment. All experiments are reviewed for hazard potential. Possible hazards include the post-irradiation radioactivity, heat generation, reactivity effects, explosivity, corrosion, toxicity, and susceptibility to radiolytic decomposition.

Once the hazards are identified, appropriate methods for monitoring and control are specified. For example, the following approach might be taken to minimize radiation exposure from a beam port: careful design of shielding, employment of a beam stop, and use of a warning device that would alert anyone who approached the beam itself. Similarly, for an automatic transfer tube, monitoring and control might be effected by installing a radiation monitor that reads out remotely and by limiting access to the sample eject station in accordance with provisions for a high radiation area. Most experiments do not generate heat. However, if alpha decay or fission is involved, it might be necessary to monitor the sample for temperature through use of a thermocouple or other device. Very few experiments involve significant reactivity effects. If one does, then its interaction with the MITR could be limited by restricting sample size or even by installing a special experiment scram. Experiments for which hazard control is not satisfactory are not permitted.

All methods for the monitoring and control of experiments are verified for operability prior to use. In addition, they may be subject to periodic surveillance and testing.

Written guidance is provided to prospective experimenters to assist them with the design of their equipment and/or content of their work. This includes information on the calculation of activity, the estimation of heat generation, and the sensitivity of materials to radiation. Each experimenter (or team) is responsible for the preparation of a description of the planned experiment, a safety evaluation, and procedures for conduct of the work. This material is then reviewed independently by staff of both the Reactor Radiation Protection Office and Reactor Operations. The latter also verify that there is no unreviewed safety question. A safety review and/or an irradiation request approval is issued that summarizes the evaluation of these two groups. If the proposed experiment is not within the envelope of previously approved uses of the reactor, then the safety review is provided to the MIT Committee on Reactor Safeguards for approval.

All experimenters are required to undergo training on radiation safety and on conduct within the restricted area.

10.2 Experimental Facilities

10.2.1 Fission Converter Medical Irradiation Room

A medical irradiation room is located on the main floor of the reactor building. Thermal neutrons from the MITR's graphite reflector impact on a fission converter plate. Fission neutrons from the plate are then filtered to produce an epithermal beam with superior dose/depth penetration characteristics. The beam is scheduled to become operational in the latter part of the year 2000. It will be used in conjunction with MIT's ongoing patient trials of neutron capture therapy. A separate safety analysis was prepared of the fission converter and submitted to the U. S. Nuclear Regulatory Commission [10-1]. Accordingly, it is not discussed further here.

10.2.2 Basement Medical Irradiation Room

The original MITR medical therapy room is located directly under the reactor. A penetration in the room's ceiling allows entry of a neutron beam. This facility is currently used to support patient trials of neutron capture therapy for the treatment of glioblastoma multiforme (brain tumor) and metastasized melanoma. Other uses such as radiography, track etch, and medical research are also possible.

The medical therapy room is shielded by at least 4.5 feet of concrete except for a transparent stepped viewing window (4.5 feet of mineral oil and glass) and a sliding door (7 inches of lead and 2 inches of steel). This shielding allows personnel who are attending a patient to remain in the work area external to the room while the beam is on; the average radiation level is less than 0.5 mrem/hour. The beam itself is controlled by four shutters that are operated from the therapy control panel that is mounted on the exterior of the medical facility. The four shutters are the D_2O blister tank, the H_2O shutter tank, the boral shutter, and the lead shutter.

The D_2O blister tank is contained within the reflector tank. The H_2O shutter, a circular aluminum tank positioned 6 inches below the reflector tank and offset 3-1/2 inches from the reactor's vertical centerline, is 20 inches deep. The boral shutter is a 1/2-inch movable plate of boral that is operated by a pneumatic cylinder and controlled by electrical solenoid valves. The lead shutter is a 9-inch thick slab of lead supported on a small wheeled dolly in the medical therapy room ceiling that is operated by a pneumatic cylinder and controlled by electrical solenoid valves. When the lead shutter is open, a bismuth plug is positioned in the beam.

A viewing window that is filled with mineral oil allows observation of the room from the outside control area. Electrical leads and other utilities may be brought into the room through openings in the shield wall below the control panel. The room is outfitted with a hydraulically-elevated operating table, closed-circuit TV for patient monitoring, and radiation monitors.

10.2.2.1 Shutter Operation and Control

The D_2O shutter is operated as an auxiliary system of the heavy water reflector's level controller. The valves and piping are shown in Figure 5-2. Opening and closing is achieved via the solenoid valves DV-51 and DV-47. The normal configuration is for the shutter to be closed (full of D_2O). There is no flow through the D_2O shutter tank. Rather an overflow pipe is filled to the level of the heavy water reflector. Heat removal is by conduction to the heavy water reflector.

The valves and piping for operation of the H_2O shutter are shown in Figure 5-1. A portion of the flow (~10%) from the auxiliary pump, MM-2, is directed through solenoid valve BV-21 to the shutter tank. The tank overflows to the primary storage tank. To open the shutter, BV-21 is closed and the contents of the tank are pumped by pump BM-1 to a shielded storage tank. To fill the shutter, BM-1 is stopped and BV-21 is restored to its normal open position. Solenoid valve BV-30 is closed for several minutes so that the auxiliary pump does not also pump extra water to the shutter while it is refilling.

The lead and boral shutters are normally operated by pneumatic cylinders. Low air pressure causes the shutters to be closed automatically. An accumulator tank provides a reserve air supply. These shutters can also be closed manually from a position adjacent to the medical therapy control panel. Because of the manual closing method for the lead or boral shutters, the reserve air supply, and the fail-safe closing of the water shutter, the system is deemed adequate for shutter closure, and therefore shutter operation is not a part of the emergency power system.

The shutters are operated from the medical therapy control panel which is outside the room. This panel contains a key switch that must be turned on in order to provide power to the controls. Also located on this panel are level indicators for the shutters, a button to open the shield door at the medical room entrance, a pilot light indicating when the air pressure for the pneumatic systems is normal, and a button which scrams the reactor. The control circuits are set up so that if the shield door to the facility is opened, the lead, boral, and H_2O shutters automatically close to cut off the beam.

There is another control panel located inside the medical room. Its controls consist of shutter position indicator lights and "close" buttons for each of the four shutters. The shutters cannot be opened from inside the room.

10.2.2.2 Radiation Monitors

There are two radiation monitors associated with the basement medical therapy room. Both are wall-mounted GM units. The first unit is part of the reactor's area radiation monitoring system which is described in Section 7.7.1 of this report. That system's purpose is to provide warning to personnel in the reactor building of elevated radiation levels. The second unit is dedicated to the performance of patient therapies. Its purpose is to alert attending personnel who work in the medical room of elevated radiation levels. If such levels exist, the following indications are given:

a) A warning lamp at the monitor in the therapy room (visible at the medical control panel);

b) An audible signal in the therapy room; and

c) An audible signal at the therapy control panel.

This unit's audible alarm is cut out during an actual therapy so as not to annoy the patient. The alarm is interlocked so that it is automatically reactivated upon opening of the therapy room door.

Both radiation monitors are transferred to emergency power upon loss of normal electrical power.

10.2.2.3 Access Door

The shield door at the entrance to the basement medical room is motor driven and is normally controlled from push-button stations located inside and outside the door. In an emergency, the door can be opened manually with a wall-mounted winch. The door can also be opened by a button at the main therapy control panel. Lamps on the control room annunciator panel and on the therapy control panel signal when the shield door is not closed. This door is interlocked with the facility shutters to close the lead, boral, and H_2O shutters whenever it is opened. Indication of the door's position is provided in the reactor control room.

10.2.3 Beam Ports

Beam ports penetrate the reactor biological shield to provide neutron beams for both experiments and irradiation facilities as well as locations for nuclear instruments. The eleven main radial beam ports are located so that their source of neutrons is several inches below the reactor core so that any neutron entering a port must have undergone at least one thermalizing collision prior to entry. This port location, which is sixteen inches below the center line of the MITR-III fuel corresponds to the thermal flux peak below the core. It is taken as a zero reference in the following discussions. Table 10-2 is a summary of the beam ports.

Many of the beam ports are aligned with reentrant thimbles in the heavy water reflector tank. This is done to maximize the flux available to the experimenters. Reentrant thimbles pass under the light water core tank. However, they do not penetrate that tank and hence failure of a thimble can not cause a loss of primary coolant.

Table 10-2

Beam Port Summary

Port	Description	Shutter	Reference Height
12SH1	12-inch radial port	yes	0
6SH1	6-inch radial port	yes	0
6SH2	6-inch radial port	yes	0
6SH3	6-inch radial port	yes	0
6SH4	6-inch radial port	yes	0
4DH1	4-inch radial port	no	0
4DH2	4-inch radial port	no	0
4DH3	4-inch radial port	no	0
4DH4	4-inch radial port	no	0
4DH5	4-inch radial port	no	0
4DH6	4-inch radial port	no	0
6RH1	6-inch semi-radial port	no	+ 9 inches
6RH2	6-inch semi-radial port	no	+ 9 inches
6TH1	6-inch through port	no	+ 12 inches
6TH2	other end of 6TH1	no	+ 12 inches
4TH1	4-inch through port	no	-14 inches
4TH2	4-inch through port	no	-24 inches
4TH3	other end of 4TH1	no	-14 inches
4TH4	other end of 4TH2	no	-24 inches
4IH1	4-inch tangential port	no	-18 inches
4IH2	4-inch tangential port	no	-28 inches
4IH3	4-inch tangential port	no	-18 inches
4IH4	4-inch tangential port	no	-28 inches

The principal hazard associated with the beam ports is radiation from the beam itself. Protection is provided by the use of beam catchers and barriers that alarm if penetrated so as to warn of proximity to the beam. These barriers may be physical or reflected light (lasers) or of some other form. The production of Ar-41 in beam ports is suppressed through the use of purge gases where appropriate. Beam port experiments can not affect reactivity.

10.2.3.1 Main Radial Beam Ports

The eleven main beam ports are arranged on radii of the reactor tank with their centerline at a reference height of zero so that they penetrate the high flux region below the reactor core. A typical 6-inch port is shown in Figure 10-2. The port consists of a port box, a stainless steel port sleeve buried in the concrete biological shield, and a type 1100 aluminum port liner bolted to the sleeve and extending into the graphite reflector to the edge of the heavy water reflector tank. Port liners which mate with a reentrant thimble may be installed for special experiments. This will extend the facility into the high flux region near the core.

The 6- and 12-inch ports are equipped with vertical shutters that are normally in the lowered or open position. When raised, the shutters will insert about one foot of lead into the neutron beam path. The shutters are operated from the reactor top.

Sealing a beam port that is not in use is a three step process. First, a plug is placed in the port. The sleeve's inner surface is machined to close tolerances so that the port plug will fit tightly thereby minimizing radiation streaming. In addition, the sleeve and plug are stepped so as to preclude any further radiation streaming. Second, gas seals are made by bolting a ring against the aluminum flange of the liner so as to force the soft aluminum on the back of the flange into circumferential grooves machined in the stepped portion of the sleeve. Finally, a gasketed cover can be bolted over the beam port's opening.

The 12-inch port, 12SH1, has a liner that is 12-11/16 inch ID with a side wall thickness of 3/16 inch and an end plate thickness facing the reactor tank of 1/16 inch. The 6-inch ports, 6SH1-6SH4, have liners that are 6-5/8 inch ID with side walls 1/8 inch thick and an end plate thickness of 1/16 inch. The 4-inch ports, 4DH1-4DH6, have liners that have a 4-1/2 inch ID with a 0.094-inch wall and a 1/16-inch thick end plate. Liners extend to within approximately 1 inch of the heavy water reflector tank wall.

The distance from the inner face of the port box to the bolt-up ring is about 50 inches. The distance from the bolt-up ring to the inside end of the liner is 31-3/4 inches.

10.2.3.2 Six-Inch Semi-Radial Ports

Port 6RH1 is on the north side of the reactor with a reference height of +9 inches. It is aimed off-center to the reactor tank by several inches. 6RH2 is in a corresponding position on the south side of the reactor. The liners for these ports are similar to those of the main radial ports, except the 6RH1 liner extends into a reentrant thimble to accommodate a permanently-installed pneumatic tube assembly. The liner flanges are bolted directly to the radial port housing. The distance from the flange to the reflector tank wall is about 60 inches. A 13-inch reentrant thimble welded to the reflector tank then extends into the high flux region.

10.2.3.3 Six-Inch Through Port

The 6-inch through port is designated 6TH1 on the north or personnel lock side of the reactor floor and 6TH2 on the south side. A line drawn through the two facilities would pass within 1 inch of the reflector tank on the 12-inch port side of the reactor. Furthermore, this line would be perpendicular to the 12-inch port. The centerline reference height for these ports is +12 inches. The distance between the port liner bolt-up rings is about 115 inches of which about 72 inches is in the graphite region. The distance from the bolt-up ring to the port box is about 50 inches. The liner is 6-5/8 ± 1/32 inch ID with a 1/8-inch wall. A gas seal is made at one end by the usual circumferential grooves. The opposite end is smooth pipe without a flange. A seal ring fits over the end and is bolted to the port sleeve, thereby compressing a rubber O-ring against the liner and the port sleeve.

10.2.3.4 Four-Inch Through Ports

The 4-inch through ports on the north side of the reactor are designated 4TH1 and 4TH2 while those on the south side are 4TH3 and 4TH4. 4TH1 is located above 4TH2. The same relation applies to 4TH3 and 4TH4. A line drawn through either 4TH1 and 4TH3 or though 4TH2 and 4TH4 would pass within 1 inch of the reflector tank on the fission converter side of the reactor. In addition, this line would be perpendicular to the centerline of the 12-inch port. The upper port centerline reference height is -14 inches. The lower reference height is -24 inches. The distance between bolt rings is about 115 inches of which about 72 inches is in the graphite region. The distance from ring to port box is about 50 inches. The liner is 4-1/2 ± 1/32 inch ID, 0.094-inch wall, with gas seals similar to those of the 6-inch ports.

10.2.3.5 Horizontal Instrument Ports

The four 4-inch horizontal instrument ports are spaced 1 inch from the reactor tank with centerline parallel to the centerline of the 12-inch port. All are on the 12-inch port side of the reactor. Port 4IH1 is on the north with a centerline reference height of -18 inches while 4IH3 is on the south at the same elevation. 4IH2 on the north and 4IH4 on the south are directly below their counterparts with a reference height of -28 inches. 4IH1 and 4IH2 are served by a common port box as are 4IH3 and 4IH4.

10.2.4 Automatic Transfer

10.2.4.1 High Flux Pneumatic Tube

6RH1 has a pneumatic tube assembly, 2PH1, permanently installed in a straight hole with exterior shielding. This facility allows a sample to be injected rapidly into the reflector tank reentrant thimble for irradiation. The sample is placed in a cylinder (rabbit) that has a 1-3/8 inch ID and is 7-1/4 inches long (6-1/4 inches long inside). These cylinders are normally made of polyethylene. However, other materials could be used. The rabbit is then placed in the 2-1/4 inch pneumatic tube for insertion. The design of this facility is similar to those of the one-inch pneumatic tubes except that 2PH1 samples are exposed to a higher flux because the facility terminates in a reentrant thimble. Also, larger samples can be accommodated. The send/receive station for this facility is located inside the shielded cell in the secondary chemistry area. Operation of the 2PH1 facility is similar to that of the 1-inch pneumatic tubes. (See Section 10.2.4.2 of this report.)

The principal hazards associated with this facility are Ar-41 production and sample activity. The former is minimized by maintaining a CO_2 gas purge when the tube is not in use. The latter depends on the nature of the sample. All samples exit to a sample processing area to which entry is restricted in accordance with the definition of a high radiation area. This area is equipped with a radiation monitor that reads out remotely. This allows radiation readings to be taken remotely before handling a sample. Samples that are inserted into this facility may have a very minor reactivity effect, less than a few millibeta (5 millibeta at worst). This is not of significance to reactor operation and it is well below the movable reactivity limit for samples.

10.2.4.2 Pneumatic Tube System (1PH1, 1PH2, 1PH3, 1PH4)

Two chemistry areas are located in the basement area of the reactor. Each is furnished with two pneumatic tubes, which allows samples to be injected rapidly into the graphite reflector region of the reactor for irradiation. The sample is placed in a 1-inch ID by 2-5/16 inch long polyethylene cylinder called a rabbit. Polyethylene end-caps are threaded into each end of the cylinder and the rabbit is then placed in the 1-1/2 inch diameter pneumatic tube. (Note: Materials other than polyethylene may be used.)

Each pneumatic tube consists of two concentric tubes which terminate in the graphite region about 1 inch from the reflector tank. The inner tube is 1-1/2 inch by 20 gauge 1100 aluminum tubing and is perforated at the in-reactor end. It butts against a 1/4-inch thick cap on the outer tube. The sample rides in and out through the inner tube.

Application of a vacuum to the inner or outer end of the sample tube allows the rabbit to be either inserted or ejected by means of differential air pressure. The vacuum provided by the facility's pneumatic blower is applied to the tubes by means of a bank of solenoid valves. When a rabbit is inserted, the solenoids operate in such a manner that a vacuum is channeled between the inner and outer tubes. Given the perforations in the end of the inner tube, the vacuum will be applied to the in-reactor end of the sample tube. At the same time, another solenoid opens a valve that applies atmospheric pressure to the outer end of the sample tube. The resulting pressure differential then carries the rabbit rapidly through the sample tube. The operation is reversed by a second pair of solenoid valves in order to eject a rabbit.

The in-graphite portions of the pneumatic tubes are horizontal with their centerlines perpendicular to that of the 12-inch port and located on the same side of the core tank as this port. The horizontal distance from the centerline of a tube to the vertical centerline of the reactor tank is 25-3/4 inches. Tube 1PH1 enters from the north while 1PH2 enters from the south directly below the 6-inch through port. Tube 1PH3 enters from the north and 1PH4 from the south. Both are located 3-1/2 inches lower than 1PH1 and 1PH2. The ends of the tubes terminate 1 inch on either side of the centerline of the 12-inch port. Tubes 1PH2 and 1PH4 are operated from the primary chemistry area while 1PH1 and 1PH3 are operated from the secondary chemistry area.

An extension of the pneumatic tube system enables samples to be transmitted directly to the nuclear chemistry laboratory that is located in NW13, an adjoining building. The two systems are not directly connected and the sample must pass clear of the reactor's pneumatic tube before transferring to the other system. A sample inserted in 1PH1 may be ejected into either the shielded cell in the chemistry area or into the tube leading to the nuclear chemistry laboratory. In order to transfer samples to the chemistry lab, the appropriate interlocks must be satisfied. Once ejected from 1PH1, a sample pauses in the hot cell in order to ensure that its activity is less than the preset value allowed by the hot cell detector. The transfer station then aligns automatically with the pneumatic tube so that the sample may be sent to NW13. A separate pneumatic blower, located in the nuclear chemistry building, draws the rabbit through a tube that penetrates the containment shell, passes underground through the Nuclear Reactor Laboratory building, rises up behind a shield wall, and exits at the receiving station in the laboratory. The total elapsed time from ejection to arrival at the receiving station is approximately 5 seconds. This system enables experimenters to study short-lived radionuclides.

Although 1PH1 is the only pneumatic tube used for this system, samples irradiated in other facilities can be transferred to the nuclear chemistry lab by placing the 1-inch rabbit containing the sample in the hot cell send station and satisfying the appropriate interlocks. The automatic valve can be opened and the sample transferred. Appropriate interlocks are installed so that a rabbit cannot be transmitted from the reactor building unless it is so intended. An air-operated quick closing valve seals the reactor containment when the system is not in use. A manually-operated valve is also installed for additional protection.

Samples can be inserted from either the secondary chemistry area or from the NW13 hot lab. The latter is accomplished mechanically by reversing the process that transfers samples from the reactor to NW13. Rigid procedural controls ensure that only approved materials are inserted and that licensed personnel control the actual insertions.

The production of Ar-41 in the 1-inch pneumatic tubes is suppressed by the use of a CO_2 purge gas whenever a tube is not in use. Personnel exposure is controlled as described above. Namely, all samples exit to a shielded sample processing area so that radiation levels can be checked before the sample is handled. There is no reactivity effect associated with operation of any 1-inch tube.

10.2.5 Sample Processing Areas

Two sample processing areas have been provided for use in examining irradiated samples. The first is referred to as the "primary chemistry area." It consists of an entire room that is located in the basement next to the control room. The second, which services irradiations performed in 1PH1 and 2PH1, is an L-shaped unit located in the basement near the far door of the equipment room. It is referred to as the "secondary chemistry area shielded enclosure."

10.2.5.1 Primary Chemistry Area

The primary chemistry area contains the send/receive station for the 1PH2/1PH4 pneumatic tubes. Samples exiting these tubes are ejected into a small enclosure made of lead bricks. A ventilation duct located at the top of this enclosure ensures that any gaseous or particulate activity is drawn into the exhaust ventilation. In addition, this chemistry area contains a ventilated hood. The hood and the pneumatic tube's exit station share a common blower. Full suction on the hood can be obtained only by closing the damper to the pneumatic tube's send/receive station.

10.2.5.2 Secondary Chemistry Area

The secondary chemistry area is used principally to work with samples that have been irradiated in either the 1PH1 or the 2PH1 pneumatic tubes. All such samples are either stored in lead-lined holders within the L-shaped lead enclosure or transferred to the radiochemistry laboratory in building NW13 via the 1PH1 pneumatic tube.

The enclosure is constructed of two rows of lead bricks that are staggered to prevent radiation streaming. Entry is provided through a door which is kept locked because the interior of the enclosure may be a high radiation area. The enclosure is equipped with two manipulators and a leaded-glass window so that the operator can work with samples inside the lead walls while being shielded from the radiation. The enclosure contains both a radiation monitor which reads out externally and a mechanical device for opening or closing the sample containers. Air pressure within the enclosure is maintained slightly negative relative to that of the rest of the building. This ensures that any particulate or gaseous activity released while opening sample containers will be drawn into the exhaust ventilation.

10.2.6 In-Reflector (Graphite) Irradiation Facilities

Six vertical thimbles located in the graphite reflector slant downward from the lower annular ring to a point about 27 to 28 inches below the centerline of the fuel. The bottom ends of these thimbles are about 1 inch from the D_2O reflector tank. The exceptions are 3GV3 and 3GV4 which are several inches away.

Some of the vertical thimbles are used for sample irradiation facilities. Gamma heating within the facilities necessitates the existence of a water-cooling system to avoid sample damage. The cooling water is provided by the shield coolant system. Temperature-resistant materials may be irradiated in the non-cooled vertical holes. Some of the vertical thimbles are used for nuclear instruments.

Samples may be inserted into or removed from the neutron flux while the reactor is at power. However, they can not be removed from the facility until the reactor has been shut down or, in some cases, until the power level has been reduced.

A slight CO_2 purge is maintained to the sample thimbles while the reactor is operating. The blanket of CO_2 displaces the air in the thimble which helps to reduce the amount of Ar-41 produced. Equally important, the CO_2 purge prevents the formation of nitric acid. Gamma radiation will cause the nitrogen and oxygen in air to combine to form nitrous oxide, a red-colored gas more commonly known as laughing gas. Nitric acid, a powerful corrosive, results when nitrous oxide combines with water. Given that 3GV facilities may be water-cooled, there may be condensed moisture on the insides of the thimbles. Hence, were air to be present, the environment would be prone to acid production.

There is little or no reactivity effect associated with the use of any of these facilities.

10.2.7 In-Core Sample Assemblies

In-core sample assemblies (ICSAs) may occupy one or more fuel element positions. The fuel loading of the MITR is quite flexible in that there are many possible loading patterns that satisfy the shutdown margin and thermal-hydraulic criteria enumerated in Sections 4.5.3.3 and 4.6 of this report. Accordingly, there may be as many as four ICSAs at a time. However, none are permanently installed and each facility's design depends on the particular experiment that is to be conducted. Individual safety evaluations are prepared for each new type of ICSA. ICSAs have been used for sample irradiations, the study of water chemistry under conditions replicating those that exist in both pressurized and boiling water reactors, and the evaluation of crack propagation in steel alloys.

Each assembly consists of an aluminum thimble that may be as much as two inches in outer diameter. This thimble is positioned inside an aluminum jacket whose outside geometry is the same as that of a standard fuel element so that it will be held securely by the hold-down grid in the same manner as are the fuel elements. This design also ensures that ICSAs do not alter the distribution of coolant to the fuel elements.

Penetrations are provided through the upper shielding to provide for support services such as instrumentation. All such penetrations are normally stepped or slanted to reduce radiation streaming.

ICSAs are required to conform to certain design criteria. These are:

a) Cross-sectional area must be less than 16 square inches.

b) No fueled loops are permitted.

c) Materials of construction are limited to those permitted for use in the primary system.

d) Adequate cooling must be provided to remove both the heat from gamma ray attenuation and that from any electric heaters that are used to control sample temperatures.

e) Reactivity effects are less than the allowed amount for the category of the experiment (movable, non-secured, secured - Sections 10.3.2.6(b) and (c) of this report).

f) Materials contained within the ICSAs (and hence isolated from the primary coolant) are within prescribed limits for toxicity, explosivity, radiolytic decomposition, etc.

Each new type of ICSA is subject to a safety evaluation prepared by the experimenter, a safety review prepared by Reactor Operations and Reactor Radiation Protection, and review by the MIT Committee on Reactor Safeguards. (<u>Note</u>: See Appendix A to Chapter 13 of this report for an example.)

10.2.8 Reactor Floor Hot Cell

The reactor floor hot cell provides protection for operators and experimenters working with material that has been irradiated in any of the in-core facilities or has been received from another facility. This hot cell consists of two subcells that can be used for storage of irradiated materials, packaging of samples, and gross visual examination of specimens. An adjoining shielded cave contains specialized equipment, including a closed circuit television monitor, for in-depth analysis of metallurgical specimens.

The hot cell, exclusive of the cave, is 16 feet wide, 7 feet deep, and 12 feet high. This provides space for the two subcells, each approximately 5 feet wide, 4 feet deep, and 10.5 feet high. The subcells are separated by a shielding wall. The front wall is constructed of an 18-inch thick slab made of dense concrete and iron punchings. The shielding factor for this slab for 2.0 MeV gamma rays is 1250. The top, sides, back, and divider are also 18 inches thick. However, these are made of ordinary concrete for which the shielding factor is approximately 70. Further shielding for the back of the cells is provided by the 2-foot thick containment wall. The area beneath the hot cell, which is the reactor control room, is shielded by the 36-inch concrete floor which has a shielding factor of 5000. The hot cell puts an average loading of 1300 pounds per square foot on the main floor of the reactor building. This is well below the design limit of 2000 pounds per square foot. Each subcell has access ports for remote manipulation and a viewing window constructed of three 6-inch thick panes of leaded glass. These windows have shielding factors of approximately 140,000. The panes are stepped with the inner one having the largest dimensions. This design prevents radiation streaming. The inner wall surfaces have been sealed and a metal pan placed at the bottom of each cell to aid in decontamination. The sealant is a flame-resistant epoxy.

10.2.8.1 Ventilation

The cells are kept at a negative pressure relative to the rest of the containment building. This ensures that any airborne activity will be prevented from escaping. The negative pressure is achieved with an air flow of about 100 cfm per cell to the containment building ventilation system. Both inlet and outlet air passes through 2-inch roughing filters. The outlet air is then ducted to a fire-resistant 6-inch HEPA filter, a blower, and finally into the building exhaust before the radiation monitors, holdup plenum, and filters. The hot cell blower is electrically interlocked with the building exhaust to shut off when the main ventilation is off (the same as other auxiliary blowers). Ventilation is monitored by means of a manometer on each cell, and procedures specify the minimum differential pressure required for cell use. A local visual alarm notifies the hot cell operator if the differential pressure is too low.

10.2.8.2 Access Control and Radiation Monitoring

The interior of the reactor floor hot cell, including the cave, may be a high radiation area depending on what samples are in this facility. Access to the two subcells is normally through one of four 6-inch roof ports. These ports are normally closed with a stepped concrete plug which can be moved only with the reactor crane. Inasmuch as the crane's controls are kept locked unless it is in use by an authorized NRL staff member, access to the two subcells is restricted to authorized users. Access to the cave is controlled by means of a locked gate.

Each cell has an internally-mounted radiation monitor that is capable of reading levels to 1000 R/hour. A reactor floor area radiation monitor that is mounted external to the hot cell is used by the console operator to monitor radiation levels on the reactor floor in the vicinity of the hot cell.

10.2.8.3 Fire Protection

Fire extinguishers are mounted external to each subcell so that the cell operator can flood the cell from his working position. Each subcell is equipped with a thermal fire detector that will, in the event of a fire, trip the cell's ventilation, sound a local alarm, and activate a smoke detector alarm in the control room. The potential for fire is minimized by limiting the amounts of flammable materials, such as paper used in reducing contamination. Decontamination of the hot cell interiors is done with nonflammable detergents, except that small amounts of solvents such as acetone may be needed to clean the manipulators. Rags and similar materials are stored in metal containers with self-closing lids.

10.2.9 Gamma Irradiation Facility

An 8-foot diameter by 21-foot deep steel-lined concrete tank in the reactor building's basement serves as a storage pool for spent fuel and spent control blades. The pool is filled with demineralized water which serves both to remove decay heat from the fuel and to shield personnel from the gamma radiation produced by the fuel elements. A filter and demineralizing system runs continuously thereby preventing the accumulation of particulate matter in the tank. Spent fuel elements, which are stored in cadmium-lined racks in the bottom of the tank, produce a gamma field that can be used for irradiations.

10.2.10 Closed-Loop Digital Control

The MITR has been used since 1981 to demonstrate techniques for the closed-loop digital control of nuclear reactors. Areas that have been demonstrated on-line include signal validation both without and with analytic redundancy, proportional-integral-derivative control, the reactivity constraint approach, rule-based control, and period-generated control. Several of these techniques have subsequently been tested at a reactor operated by one of the national laboratories.

Digital control is no different than analog or manual control in terms of the potential challenge that can occur to the reactor protection system. All types of control involve manipulation of a control device with the worst-case scenario being a continuous (ramp) insertion of reactivity. Protection against this type of accident is provided for the MITR by limiting the number of control devices that may be withdrawn at a time and by specifying an upper limit on the rate of reactivity insertion. (See Section 13.2.2.2 of this report.) These restrictions apply to all modes of control including digital. The safety of an automatic controller (analog or digital) can be further assured by limiting the reactivity worth of the associated control device. This is a standard approach, one that has been utilized since the inception of the nuclear industry. For example, the worth of the

MITR's regulating rod is limited to 0.5% $\Delta K/K$ because this amount of reactivity is well within the inherent shutdown capability of the core by the negative reactivity effects of temperature and voids.

A restriction on the magnitude of the reactivity of the associated absorber is the best that one can achieve to ensure the safety of a traditional analog controller. However, digital control allows improved safety. Specifically, one can design a digital controller to incorporate the concept of "feasibility of control" [10-2]. As applied to a nuclear reactor, the term "feasible to control" means that a controller shall be capable of transferring the reactor from a given power level and rate of change of power to a desired, steady-state power level without overshoot, or conversely, undershoot. (Note: If a deviation band is specified about the desired power level, then the term "without overshoot" means that there will be no overshoot beyond the permitted deviation.) This property can be attained by designing the controller so that the net reactivity inserted by that controller can be offset by the insertion of the associated control device before the limiting power level is attained. This concept is the "reactivity constraint approach" [10-2]. Its basis may be found by examining the equations of reactor dynamics and in particular the dynamic period equation, which shows that the rate of change of reactor power depends on both the net reactivity and the rate of change of reactivity. This fact, coupled with the fact that the rate of change of reactivity is, under non-scram conditions, finite means that adjustments in the reactivity must be preplanned if power overshoots are to be avoided. Digital controllers can be designed to achieve this. Specifically, the reactor period can be made rapidly infinite if the total reactivity, both that added directly by the control mechanisms and that present indirectly from feedback effects, is maintained less than the maximum available rate of change of reactivity divided by the effective, multi-group decay parameter. Physically, if the reactivity is so constrained, then, by reversal of the direction of motion of the specified control mechanism, it will be possible to negate the effect of the reactivity present and make the period infinite at any time during the transient. This condition, which is referred to as an "absolute reactivity constraint," is unnecessarily restrictive. A less stringent constraint may be used that specifies that there be sufficient time available to eliminate whatever reactivity is present beyond the amount that can be immediately negated by reversal of direction of the designated control mechanism before the desired power level is attained. This condition is the "sufficient reactivity constraint." Its function in a controller is to review the decision of whatever control law is being used and, if necessary, override that decision. Provided that the net reactivity is always restricted to that permitted by the sufficient constraint, it should always be possible to halt a power increase at the desired termination point by merely reversing the direction of absorber travel. Therefore, adherence to this constraint means that no automatic control action should ever result in a challenge to the nuclear safety system.

Digital controllers that incorporated the sufficient reactivity constraint concept were demonstrated on the MITR with the regulating rod in 1983. These demonstrations were repeated in the mid-1980s with one of the MITR's shim blades. The approach has also been used on the Annular Core Research Reactor that is operated by the Sandia National Laboratory [10-3].

The design of a controller that incorporates the reactivity constraint approach allows the study of many different types of critical laws. At the MITR, such studies have included rule-based control, which is a form of artificial intelligence, and period-generated control, which is a model-based method similar to the computed-torque technique that is used in robotics. These studies are performed by using a two-tier structure. The lower or first level of the controller would use the new control strategy to be tested. That level would be allowed to control the reactor power at some fraction of full power. The decisions made at this first level would be reviewed by a second tier that has the property of feasibility of control. This second level would intervene if a decision at the first level could result in the attainment of a power level in excess of the allowed operating power.

10.3 Experiment Review

10.3.1 Requirements

An experimental facility is an appurtenance to the reactor that is generally used to contain and orient an experiment, as in the case of an irradiation thimble, or to provide a desired flux distribution, as in the case of a filtered beam. An experiment is any operation, hardware, or target (excluding devices such as detectors, foils, etc.), that is designed to investigate non-routine reactor characteristics or that is intended for irradiation within the core tank, or in a beam port or irradiation facility, and that is not rigidly secured to a core or shield structure so as to be a part of their design. Approval of an experimental facility and/or experiment is based on the following criteria:

a) The consequences of the planned operation and the credible failures of the facility and/or experiment are predicted to be smaller than the accidents that have been analyzed as part of the Safety Analysis Report (or appear as bases for the technical specifications),

b) An unreviewed safety question does not exist,

c) All handling procedures and the consequences of single failures are predicted not to generate radiation levels or radioactivity releases in excess of 10 CFR 20 limits for both on-site and yearly off-site averaged levels, and

d) All activities related to the experimental facility and/or experiment are in accordance with the MITR ALARA program.

The ultimate responsibility for the review and approval of an experimental facility and/or an experiment rests with the MIT Committee on Reactor Safeguards (MITRSC). The methodology observed to obtain approval of a new class of experiment is as follows:

e) The experimenter is responsible for providing a description of the proposed experiment, a statement of its objectives, and a safety evaluation. The latter includes an estimate of any radioactivity that will be produced and an assessment of any special hazards such as heat generation, reactivity, pressure buildup, etc. The experimenter is also responsible for obtaining license approval from the MIT Radiation Protection Office (as opposed to the MIT Reactor Radiation Protection Office) to possess any radionuclides that may be produced and utilized off-site. This approval would be under the MIT Byproduct Material License.

f) Both the Reactor Operations and Reactor Radiation Protection Offices conduct independent reviews of the proposed experiment. The material prepared by the experimenter is one input to this review. Others include, but are not limited to, experience with previous similar experiments, handbooks, regulatory guides, engineering expertise, etc. The result of this review is a combined safety review for which the description of the experiment and the activity calculations are verified, hazards are assessed, an unreviewed safety question determination is made, and an ALARA assessment is performed.

g) Both the experimenter's information and the safety review conducted by the Reactor Operations Staff and the Reactor Radiation Protection Office are circulated to the members of the MITRSC for review prior to a formal meeting. The membership of this Committee has broad expertise in reactor physics, thermal-hydraulics, radiation safety, reactor engineering, industrial safety, electrical systems, radiation biology, etc. The Committee then meets to discuss the experiment. One of the items that is always considered is the existence or lack thereof of an unreviewed safety question (URSQ). The MITRSC may reject the proposal, request additional information, approve it, or if an URSQ does exist, approve submission of a request for a license amendment to the U.S. Nuclear Regulatory Commission. (Note: The existence of an URSQ does not necessarily imply that a safety hazard exists. It may simply mean that the proposed experiment is not within the scope of topics that have been previously considered in the MITR's safety analysis.)

Each step in the above procedures is followed for new experimental facilities and for new uses of existing experimental facilities. However, for proposed uses of an existing facility that fall within the envelope of those uses that have already been approved, the MITRSC may delegate its authority for approval to Reactor Operations and the Reactor Radiation Protection Office.

10.3.2 Experiment Classification and Approval Activity

All proposed experimental facilities and all proposed new uses of an experimental facility are evaluated to determine if an unreviewed safety question exists. The specific criteria used for the determination are:

a) If the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or

b) If a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

c) If the margin of safety as defined in the basis for any technical specification is reduced.

Aside from the need to review for an URSQ, there is no single classification scheme that is relevant to all situations. For example, potential reactivity effects are a major concern for in-core samples but not for those run in beam ports. Conversely, whenever the reactor is operating, radiation exposure is a potential issue for beam ports but not for an in-core sample. Accordingly, classification criteria are discussed below in terms of each type of facility.

10.3.2.1 Medical Irradiation Rooms

These rooms may be used for applications such as patient trials including therapy, the irradiation of dosimeters, and radiography. Classification for the first of these uses is discussed here. For the others, the considerations listed under beam ports and for automatic/manual transfer facilities would be more relevant.

Proposed patient trials or changes to an existing trial are evaluated against the relevant MITR technical specifications, the affiliated hospital's NRC license, and the respective quality management programs. In addition, the requirements of several non-NRC agencies are addressed including those of the MIT Committee on the Use of Humans as Experimental Subjects (COUHES), the MIT Committee on Radiation Exposure to Human Subjects (COREHS), hospital institutional review boards (IRBs), and the U.S. Food and Drug Administration (FDA). The MITR staff is principally responsible for ensuring that the radiation fluence delivered to the patient is as specified in the written directive that is prepared by the physician in charge and for operation of the medical irradiation facilities so that radiation exposures to personnel attending the patient are minimized. Issues such as dose escalation, drug selection, and drug delivery are not the responsibility of the MITR staff. The approval procedure therefore requires the principal investigators, both those from MIT and the affiliated hospital, to submit jointly any request to the cognizant MIT and hospital committee (COUHES, COREHS, and the IRBs). If necessary, information is also provided to the FDA for that agency's concurrence. The same information is provided in all instances to Reactor Operations, the Reactor Radiation Protection Office, and the MITRSC. If appropriate, these organizations will also review it for approval, modification, or denial.

10.3.2.2 Beam Ports

These are most often used for experiments involving neutron scattering, prompt gamma analysis, transmission tests, and laboratory exercises for students. The principal hazard is radiation exposure from the beam. Therefore, the concerns in the design of a beam port facility are adequate shielding, the installation of a warning system that will alert users should they physically approach the beam, the use of beam stops, and the procedures that the experimenters will follow in the conduct of the experiment. These are all standard items and the MITRSC has delegated its authority for approval to Reactor Operations and the Reactor Radiation Protection Office.

10.3.2.3 Automatic Transfer Facilities

These are used primarily for neutron activation analysis. However, other uses such as isotope production (Au-198 and Dy-165 in particular) are possible. The principal hazards are Ar-41 production, radiation exposure from activated samples, internal heating of samples, and the exclusion of hazardous materials. The issues involved are all standard and the MITRSC has delegated its authority for approval to Reactor Operations and the Reactor Radiation Protection Office provided that the proposed experiment is bounded by a pre-specified envelope of conditions. These are:

a) Ar-41 production is minimized by use of a CO_2 gas purge while tubes are not in use.

b) Radiation exposure is limited by several concurrent approaches. First, experimenters are limited to the minimum activity needed for their work. Second, an estimate of the initial activity (activity at time of sample ejection to a shielded cell) is made. If undesired short-lived radionuclides are present, a mandatory decay time is imposed. Third, experimenters are trained on the proper handling of irradiated substances and written procedures are prepared. (This is under the auspices of the MIT Radiation Protection Office and the MIT Committee on Radiation Protection.) Fourth, the experiments must have approval from the MIT Committee on Radiation Protection to possess the isotopes in question.

c) Heating that results from gamma and beta ray attenuation is removed by the cooling systems that are part of the design for the transfer tubes. Samples that would entail heat production from alpha particles are prohibited unless the request is presented to the MITRSC. Thus, lithium- and boron-containing substances and isotopes with $Z > 83$ in greater than trace amounts are not to be irradiated without MITRSC involvement.

d) Certain substances may not be irradiated such as explosives in quantities greater than 25 mg TNT equivalent. Others, such as corrosive materials, may be irradiated provided that they are doubly-encapsulated.

10.3.2.4 Sample Processing Areas

These areas are utilized primarily in conjunction with the automatic transfer tubes. The limitations imposed on the use of these areas are those associated with good health physics practice including:

a) Local ventilation is provided so that any airborne radioactivity is drawn away from those handling the samples and into the filtered exhaust ducts.

b) Local shielding is provided so that the handling of samples is in accordance with ALARA.

c) If samples can create a high radiation area, provisions exist to secure the area in accordance with 10 CFR 20 requirements.

10.3.2.5 In-Reflector (Graphite) Facilities

The considerations listed in Section 10.3.2.3 apply here as well.

10.3.2.6 In-Core Sample Assemblies

Uses include isotope production, materials testing, and the evaluation of water chemistries under PWR/BWR operating conditions. The principal hazards are sample heating, hydraulic behavior, inadvertent sample or facility movement that could create a reactivity effect, and radiation exposure during sample/facility removal. There are no permanently installed ICSAs and hence the design of each is novel. Therefore, the MITRSC rarely delegates any approval authority concerning these facilities and/or the use of these facilities. The following guidelines are used for the review of ICSAs:

a) Facility Movement: In-core sample assemblies (as opposed to samples within these facilities) are designed so that they will not move during reactor operation. This is achieved by securing the ICSA in the same manner as a fuel element. Specifically, the ICSA's outer jacket fits into the lower grid plate and the upper end is held down by the upper grid plate.

b) Sample Movement: Movement of a sample within an ICSA during reactor operation could occur for two reasons. First, it might be intended to move the sample if, for example, a short-lived isotope were being produced or if it were desired to remove a sample without disrupting other users of the reactor. Second, an ICSA failure could occur in which primary coolant floods the ICSA and the resulting buoyant and hydraulic forces lift the sample. Samples are classified in terms of their potential for movement as follows:

   i) Secured - An experiment is deemed to be secured, or in a secured position, if it is held in a stationary position relative to the reactor by mechanical means or by gravity. The restraining forces must be substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces that are normal to the operating environment of the experiment, or by forces that can arise as a result of credible malfunctions.

   ii) Non-Secured - An experiment is deemed to be non-secured if it is intended that it should not move while the reactor is critical but it is held in place with less restraint than a secured experiment.

   iii) Movable - A movable experiment is one which may be inserted, removed, or manipulated while the reactor is critical.

   (Note: The above definitions are based on the use of Regulatory Guide No. 2.2, dated November 1973. The differences are that: (1) gravity is allowed as a restraining force for the secured category and (2) "non-secured" is defined as opposed to "removable," to allow for the situation where no movement is intended.)

c) Sample Reactivity - Sample reactivity is defined as either the static reactivity as defined in Regulatory Guide No. 2.2 dated November 1973 or the reactivity associated with complete removal of a sample together with filling by water of the space previously occupied by the sample, whichever is greater. The latter will, in general, be greater than the former and hence more conservative. The allowed reactivity worth of a sample varies depending on the potential for movement. Thus, individual movable samples are limited to 0.2% $\Delta K/K$, with a total for all movable samples of 0.5% $\Delta K/K$; individual non-secured experiments are limited to 0.5% $\Delta K/K$ with a total for all non-secured samples of 1.0% $\Delta K/K$; and secured are limited to 1.8% $\Delta K/K$ with no limit on the total number. These figures are chosen so as to limit the consequences of sample movement.

Accidents resulting from the insertion of reactivity are discussed in Section 13.2.2 of this report. The 0.2% ΔK/K limit for movable experiments corresponds to a 25-second period, one which can be easily controlled by the reactor operator with little effect on reactor power. The limiting value for a single non-secure experiment, 0.5% ΔK/K is set conservatively below the prompt critical value for reactivity insertion and below the minimum shutdown margin. The sum of the magnitudes of the static reactivity worths of all non-secured experiments, 1.0% ΔK/K, does not exceed the minimum shutdown margin. The total worth of all movable and non-secured experiments will not reduce the minimum shutdown margin as the shutdown margin is determined with all movable experiments in their most positive reactive states. Finally, it was determined that for operation with forced convection, a step increase of 1.8% ΔK/K would result in fuel plate temperatures that were below the clad melting temperature and significant core damage would not result.

Sample reactivities are normally estimated during design of an ICSA by calculation and confirmed by measurement upon initial installation.

d) Sample Heating - ICSAs are required to conform to the same criteria as fuel elements. Namely, nucleate boiling on the surface of the ICSA is not allowed and centerline temperatures must not approach the melting point of the sample. Heat generation in ICSA samples is caused by radiation attenuation and, in some cases, by the use of electric heaters. Heat removal could be by conduction to the primary coolant or by forced convection using an installed heat removal system. If the latter is used, high temperature alarms and/or automatic shutoffs of electric heaters would be installed if required.

e) Hydraulic Effects - The exterior of an ICSA jacket is required to conform to the same shape as a fuel element so that there will not be any excessive bypass flow.

f) Radiation Exposure - The precautions described above in the discussion of automatic transfer facilities are followed for ICSAs as well. Samples and/or facilities are withdrawn through openings in the reactor top shield lid into shielded casks.. If appropriate, these casks are equipped with a closed ventilation system to minimize airborne contamination. The samples/facilities are then transferred to the reactor floor hot cell and/or the spent fuel storage pool.

10.3.2.7 Reactor Floor Hot Cell

Use of the reactor floor hot cell is governed by several administrative criteria:

a) Samples are limited to solids and liquids. Also, if corrosion or evaporation could lead to significant release of radioactive products, separate containment of the material must be provided to limit the release.

b) Containment ventilation must be maintained when any activity is being conducted in a hot cell.

c) Proposed uses will be reviewed for good health physics practice including adequacy of ventilation and shielding as well as the capability to insert, manipulate, and remove the sample.

The MITR Operating License allows the receipt of byproduct material that has been produced at other facilities. If this material is to be utilized in the reactor floor hot cell, then in addition to the above restrictions, it should also conform to the following:

d) Solid material only in any chemical form, nuclides Z = 3 through 83.

e) Activity limits for materials activated in other reactors:

   i) 100,000 Ci total at any time,

   ii) 1000 Ci for any one sample or specimen, and

   iii) The dose rate at one meter unshielded shall not exceed 100 rads/hour.

10.3.2.8 Closed-Loop Digital Control

Digital control of the MITR is permitted subject to either of two approaches. The first is to place a limitation on the reactivity worth of the control device that is associated with the controller. This is the traditional approach that is commonly associated with analog controllers.

The second approach is to design the controller so that it incorporates the concept of feasibility of control. For the MITR, such controllers should meet the following conditions:

a) Shim blades and/or the regulating rod may be connected to a closed-loop controller provided that the overall controller is designed so that the control of reactor power will always be feasible at either the desired termination point of any transient or at the maximum allowed operating power.

b) Only one shim blade shall be withdrawn at a time.

c) The nuclear safety system shall be separate from any closed-loop controller where "separate" means that the output of an instrument used in the safety system will not be influenced by interaction with the control system. For example, a signal derived from an instrument that forms part of the safety system would not be transmitted to the control system unless first passed through an isolation device. Thus, a short circuit or other malfunction exterior to the safety system could not be passed back to it.

d) A period trip set at or longer than 20 seconds shall be operable whenever any closed-loop controller is in use. This trip shall transfer control to manual and sound an alarm.

e) The operability of the period trip is to be tested prior to use of any closed-loop controller during any week that a closed-loop controller is to be used.

f) The maximum rate of reactivity insertion shall be less than that allowed as determined by the analysis for a ramp reactivity insertion accident.

The purpose of the first two conditions is to ensure that power overshoots will be unlikely. The third condition assures that the capability of the nuclear safety system to perform its intended function will not be compromised. The fourth and fifth conditions provide an additional reliable safety factor set more conservatively than the trips associated with the nuclear safety system. The use of this period trip in effect limits the excess reactivity that can be present. The sixth condition is standard for any control mode (manual, analog, or digital).

As discussed in Sections 4.2.2.2 and 4.2.2.4 of this report, the motors that are used to drive the shim blades and regulating rod are fixed speed. For digital control studies, it is necessary to use a variable speed motor so that the rate of change of reactivity can be varied. Overspeed protection is provided through software. In addition, it is provided through hardware, such as an overspeed limiter. Overspeed trips, both software and hardware, are set so that the maximum allowed reactivity insertion rate for the MITR is not approached.

10.3.3 Administrative Controls

10.3.3.1 Training of Experimenters

Experimenters may be from MIT or from local area universities, hospitals, and/or other organizations. If from MIT, they may plan to conduct all of their work within the reactor containment building or they may wish to use irradiated material in laboratories elsewhere at MIT.

In all cases, the training received is similar, but the responsible authority is different. The possibilities are:

a) MIT and Non-MIT Experimenters Working at the MITR : These individuals are trained by the MIT Reactor Radiation Protection Office and are issued dosimetry by that office. These individuals usually do not possess radioactive material. For example, this group includes neutron scatterers and students who do laboratory exercises.

b) MIT Experimenters Working Elsewhere at MIT: These individuals are trained by the MIT Radiation Protection Office and are issued dosimetry by that office. Their authorization to possess radioactive material is from the MIT Byproduct License.

c) Non-MIT Experimenters Working Elsewhere: These individuals are trained by their own institution's Radiation Protection Office and their dosimetry is from that office. Their authorization to possess radioactive material would be from that institution's Byproduct License, a copy of which must be on file at the MITR.

10.3.3.2 Shipment of Radioactive Material

All shipments of radioactive material are made in accordance with the regulations of the U.S. Department of Transportation (DOT). DOT periodically offers training on shipping regulations and several MITR senior reactor operators have successfully completed this DOT course. Written checklists are followed to ensure that samples are properly packaged and labeled.

10.3.3.3 Facility Access

No one may use an experimental facility without the prior approval of the reactor console operator. Positive control is maintained by this individual because all facilities are either kept locked or otherwise restricted in access. Individuals who wish to have a sample irradiated or who wish to use a facility must first obtain a signed irradiation request form. These are issued by the Reactor Operation Office and, where appropriate, with the concurrence of the Reactor Radiation Protection Office. Issuance of this form ensures that:

a) The named individual has received radiation protection training and has authorization to possess the requested isotopes if any.

b) A copy of the relevant byproduct license is on file.

c) The activity of the sample (or dose rate or dose rate of beam) is within the envelope of conditions for which approval was initially granted.

d) Anticipated radiation levels, mandatory decay times, and any special precautions (use of protective clothing/tongs, etc.) are noted.

10.3.3.4 Recovery Procedures

Procedures for recovery from an accident involving an experiment are the same as for any accident involving radioactive material. These are contained in the MITR's operating procedures (normal, abnormal, and emergency). If specialized instructions are required such as for an ICSA, procedures special to that facility are prepared. These are subject to review and approval by Reactor Operations, the Reactor Radiation Protection Office, and the MITRSC.

10.3.4 Generic Safety Assessment

Generic safety assessments are performed by Reactor Operations, the Reactor Radiation Protection Office, and the MITRSC. Each group is free to review areas covered by the others. The general delineation of responsibilities are listed below.

10.3.4.1 Review by MITR Operations

Each experiment, irradiation series or change thereto is reviewed for:

a) Conformity of the experiment with MIT Reactor Safeguards Committee approvals.

b) Conformity of the proposed design and procedures to accepted practices of industrial safety.

c) Conformity of the design and procedures to the reactor design and operating procedures.

d) Adequacy of the mechanical design to perform its function. The analysis includes, as required, such topics as mechanical and thermal stresses, hydraulic and heat transfer properties, adequacy of process equipment, etc.

e) Compatibility of materials with those of reactor systems.

f) Adequacy of experiment shielding.

g) Compatibility of design and procedures with reactor and neighboring experiment designs and procedures.

h) Conformity of instrument systems tied to the reactor instrumentation criteria established in Chapter 7 of the this report.

i) Adequacy of experiment instrumentation to perform its safety functions.

j) Conformity of the electrical system design to standard industrial practice.

k) Evaluation of consequences of experiment and reactor malfunction and experiment interaction.

l) Methods of installation, removal, and preparation for shipment.

m) License authorization of off-site recipients for irradiated materials.

n) Materials to be inserted in reactor facilities as irradiations or experiments are reviewed for:

    i) Induced activity.

    ii) Physical form of the material during irradiation.

    iii) Possibility of excessive heating including effects of heating by radiation and nuclear reactions.

    iv) Reactivity, radiation, and physical damage effects of planned movements or credible accidental movements.

    v) Barriers (e.g., containers) to the release of radioactivity.

    vi) Chemical stability and chemical compatibility with the containers and the structural material of the facility.

    vii) Criticality considerations.

After initial approval, the supervisor assigned to the experiment continues to follow the experiment to verify:

o) Continued qualification of the personnel involved to carry out the approved operational procedures.

p) Compliance of experimenters to approved procedures.

q) Conformity of shipments to DOT and/or NRC requirements.

10.3.4.2 Review by MITR Radiation Protection Office

The MITR Radiation Protection Officer or a designated alternate reviews and approves each experiment, irradiation series or change thereto. This review includes:

a) Conformity of the experiment with MIT Reactor Safeguards Committee approvals.

b) Evaluation of consequences of experiment malfunction especially the possibility of release of radioactivity.

c) Adequacy of radiation shielding.

d) Adequacy of procedures for installation, operation, and removal of experiment or irradiation to prevent activity release or excessive radiation doses.

e) Materials to be inserted in reactor facilities as irradiations or experiments are reviewed for:

   i) Induced radioactivity.

   ii) Physical form of the material during irradiation.

   iii) Barriers (e.g., containers) to the release of radioactivity.

   iv) Chemical stability and chemical compatibility of the material with the containers and the structural materials of the facility.

After initial approval, the Reactor Radiation Protection Officer or a designated alternate continues to follow the experiment to establish:

f) Continued qualification of personnel involved to handle the radioactive materials or potential radiation sources.

g) Compliance of experimenters to approved procedures.

h) Conformity of shipments to DOT and/or NRC requirements.

10.3.4.3 Review by the MIT Committee on Reactor Safeguards

The Committee reviews the use of each reactor facility for irradiations and each experiment significantly different from previously approved ones. These reviews include consideration of:

a) The design of the facility or experiment and its relation to the existing reactor operation.

b) Experiment procedures.

c) Maximum reactivity, pressure, and temperature effects.

d) Inadmissible sample materials.

e) Instrumentation.

f) Radiation levels.

g) Consequences of experiment and reactor malfunction and experiment interaction.

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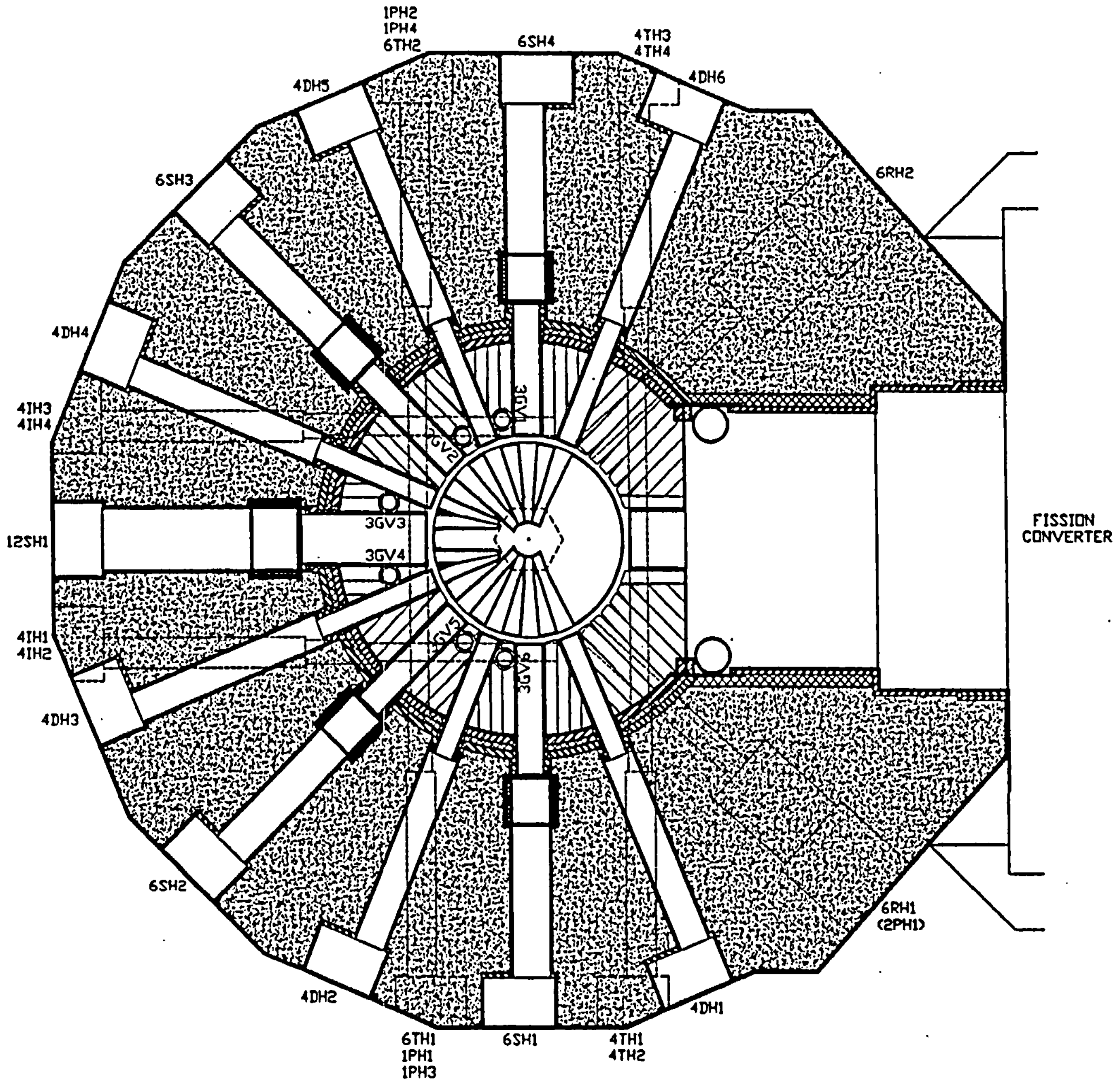

FIGURE 10-1
HORIZONTAL CROSS-SECTION

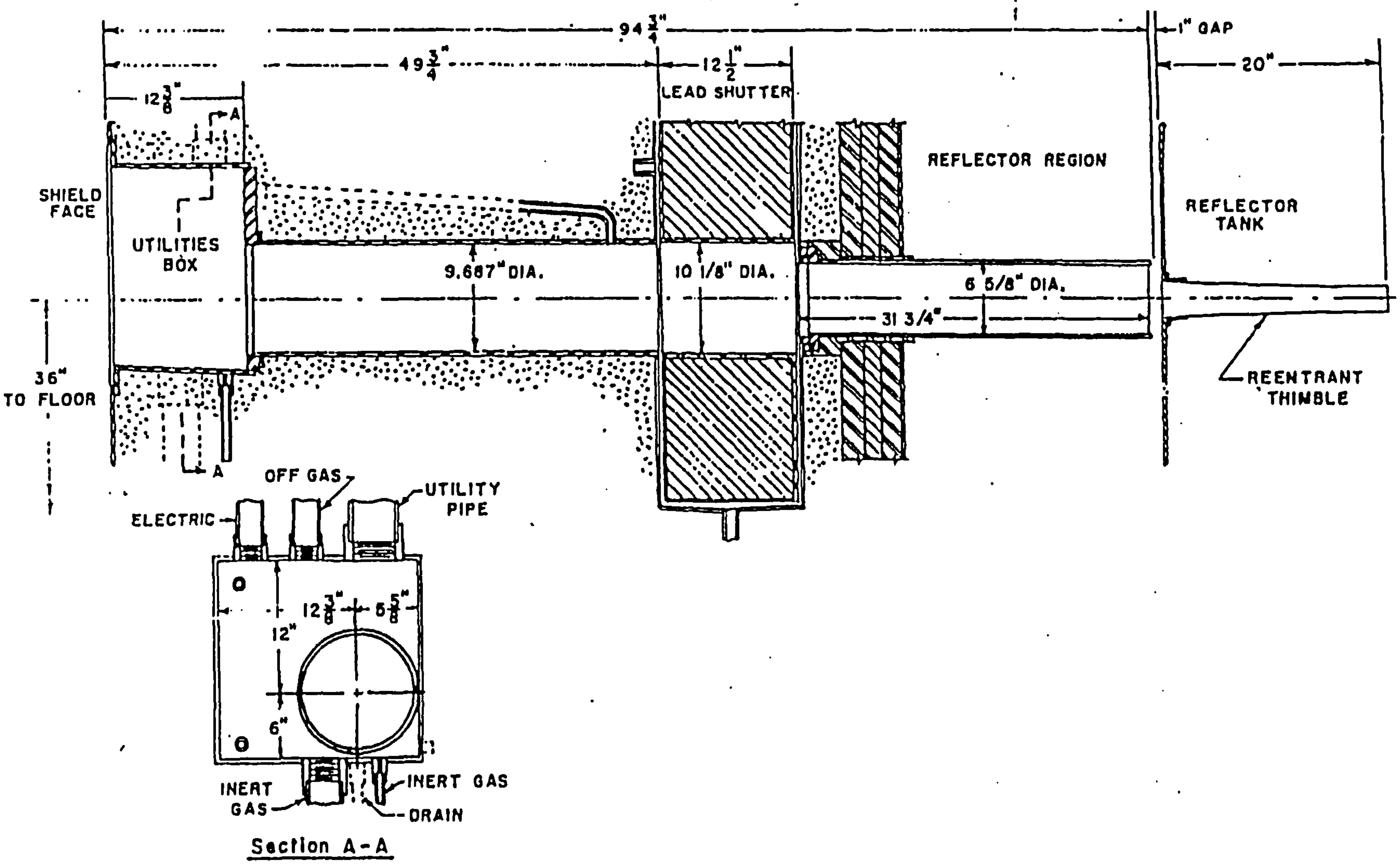

FIGURE 10-2
6" HORIZONTAL BEAM PORT

Chapter 11

Radiation Protection and Waste Management

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Chapter 11

Radiation Protection and Waste Management

11.1 Radiation Protection

11.1.1 Radiation Sources

11.1.1.1 Standard, Check, and Startup Sources

MIT utilizes standard, check, and startup sources in conjunction with the routine operation of the reactor. These sources are used to ensure operability of the effluent and area radiation monitors, to calibrate health physics instrumentation, and to provide a startup source for the reactor. Some of the sources are covered by a separate license, one for Special Nuclear Material. Sources utilized at the MIT Research Reactor (MITR) include:

a) Standards:

These sources are mainly used for the calibration and operation of health physics instrumentation to ensure reliability and accuracy. Sources include but are not limited to: µCi-level mixed and single nuclide calibration standards for analytical equipment calibration, source checking and other quality control procedures; mCi-level ^{60}Co and ^{137}Cs calibration sources for low to mid-range instrumentation; and a 10-Ci ^{137}Cs source for high range instrumentation calibrations. In addition, there are alpha and beta standards that range from 4E-4 to 10 µCi and include, for example, ^{239}Pu, ^{60}Co, and ^{99}Tc.

b) Check Sources:

The area radiation monitors, which are described in Section 7.7.1 of this report, are equipped with appropriate sources.

c) Neutron Sources

These include two 1-Ci PuBe sources and one 150-Ci SbBe source. The latter requires radioactive antimony as a precondition and is normally not active. These three sources are covered by the reactor operating license. Additional neutron sources are covered by other licenses that are held by MIT. These include both PuBe and Cf-252 sources. Additional information is given in Section 4.2.4 of this report.

Inventory and leak testing of sealed sources is performed in accordance with written procedures. The sources described are examples of the sources that are maintained. The inventory will either increase or decrease depending on the needs of the educational, research, and radiation protection programs.

11.1.1.2 Fissile and Fissionable Material

See Section 9.5.2.1 of this report.

11.1.1.3 Experimental Programs

The activity that is expected to be generated during the conduct of any experiment is estimated as part of the written review that is prepared and approved prior to the conduct of any experiment. (See Section 10.3.4 of this report.) These estimates are based on the envelope of authorized conditions, i.e., maximum flux, irradiation time, and mass. The experiment planning process also includes reviews of the adequacy of shielding, the preparation of procedures for handling the samples, and if appropriate, walkthroughs of these procedures.

In addition to generating activity through experiments conducted at the MITR, provisions exist within the MITR license for the use of byproduct material from other facilities. This is described in Section 10.3.2.7 of this report.

11.1.1.4 Airborne Radiation Sources

The principal airborne sources of radioactivity associated with operation of the MITR are Ar-41, tritium, and fission product gases.

a) Ar-41:

Air contains 1% Ar-40. Hence, when air that either is dissolved in the primary coolant or is in the vicinity of an experimental facility is exposed to a neutron flux, the Ar-40 that is naturally occurring in that air undergoes an (n,γ) reaction to Ar-41. Ar-41 has a relatively short half-life of 1.8 hours. The production of Ar-41 at the MITR is about 1 Ci/MWD. Protocols that are used to limit the production of this isotope include:

(i) Use of helium or CO_2 cover gases to eliminate air, and

(ii) Sealing of all penetrations that could allow air to circulate in regions where neutron fluxes are present.

During operation of the reactor, airborne activity concentrations of Ar-41 in the normally-occupied areas within the reactor building are typically less than 1% of the derived air concentration (DAC). However, on loss of building ventilation, significant concentrations of Ar-41 can accumulate. The source volume is approximately 5% of the entire containment volume and the resultant activity concentration within the building could approach 1E-3 μCi/ml Although this source term can be significant, it should be recognized that the DAC specified for noble gases is based on a submersion dose of semi-infinite proportions. Semi-infinite clouds for gamma-emitting radionuclides are analyzed with the assumption that the effective radius of the cloud is greater than one mean free path. For Ar-41, this radius would be at least 100 meters. The containment building radius is 35 feet. Therefore, some fraction of a semispherical infinite cloud would prevail. For small clouds (i.e., no significant scatter or attenuation), the fraction of a semispherical infinite cloud, f, can be approximated by the multiplicative value of the linear energy absorption coefficient, μ_{en}, and the radius of the cloud, r. For Ar-41 photons and an assumed maximum cloud radius of 10 m, the fraction of an infinite cloud is approximately 0.036. Therefore, instead of a 2.5 mrem/hour expected dose rate from a concentration of 1 DAC, a dose rate of approximately 0.1 mrem/hour would be observed. For other spaces within the containment having smaller dimensions, such as the control room, equipment room, setup areas, etc., this fraction would be even smaller. Regardless, monitoring for exposure from submersion dose is accomplished using dosimetry capable of monitoring the deep depth and shallow depth dose equivalents.

Any increase in reactor power would expectantly result in a linear increase in Ar-41 generation with respect to reactor power. Hence, an increase in reactor power from 5 MW to 6 MW should result in about a 20% increase in Ar-41 production.

b) Tritium:

Tritium is produced in the heavy-water reflector as the result of the radioactive neutron capture reaction [$^2H(n,\gamma)^3H$]. The heavy-water reflector is a closed system. Also, as described in Section 5.3.1.10 of this report, all flanges and pump seals in the system are monitored for moisture so that any leakage is detected in the incipient stages. Nevertheless, as discussed in Section 13.2.9.3 of this report, the consequences of a spill of heavy water have been analyzed and written procedures exist for such a contingency.

c) Fission Products:

Fission product activity can occur as the result of incipient off-gassing of reactor fuel. The nuclides that would be observed are primarily xenon, krypton, and iodine as well as their progeny cesium and rubidium. Any fission product activity would be drawn from the air space below the reactor top lid by the core purge blower and discharged to the ventilation exhaust. The latter is filtered. Radiation monitors on the core purge (used to monitor fuel performance), plenum (used to provide containment isolation on signal), and stack exhaust (used to provide effluent monitoring of building air) measure the activities of the system and provide information related to fuel performance. In addition, samples are obtained from each of these systems and are used for trending purposes.

The MITR is monitored by radiation detectors that are located on the roof tops of MIT buildings in other parts of the campus. These provide a measurement of radiation exposure from airborne releases from the routine operation of the MITR. This data, which is provided on an annual basis to the U.S. Nuclear Regulatory Commission, shows that exposures are typically less than 0.5 mrem per year. This figure should be compared to the exposure of 300 mrem per year that results from naturally-occurring radioactivity in the environment.

11.1.1.5 Liquid Radioactive Sources

Liquid radioactive sources include the primary coolant, heavy-water reflector, and shield coolant systems. Activities in these systems are minimized both by maintaining high standards of water purity and by utilizing clean-up systems.

a) Primary Coolant:

The principal nuclides are:

(i) N-16 which is the result of an (n,p) reaction on O-16,

(ii) Na-24 which is the result of an (n,α) reaction on Al-27, and

(iii) Al-28, which is the result of an (n,γ) reaction on Al-27.

The half-lives of these species are 7.13 seconds, 14.96 hours, and 2.25 minutes, respectively. Hence, none of these nuclides is long-lived and the first and third decay quite rapidly. The primary system is, with the exception of the core purge, sealed. Hence, release of these nuclides is not an issue during routine operation. Moreover, even if a spill did occur, the release would be contained within the reactor building. The

Na-24 activity concentration in the primary coolant depends on the power history and ranges from 0.2 - 0.5 μCi/ml.

b) Heavy-Water Reflector:

As discussed above in Section 11.1.1.4 of this report, the principal nuclide in the heavy-water reflector is tritium. The production rate, which follows first-order kinetics, is a function of the reactor power history. Hence, for an initial clean load of heavy water (tritium concentration of approximately 30 μCi/ml), the concentration buildup would be linear for the first few years of operation and then gradually approach a saturation value. The maximum concentration of tritium observed for this system was 3.5 Ci per liter in 1991 after 17 years of operation at the rated power of 5 MW. The near-equilibrium concentration for 6 MW operation is expected to be approximately 5 Ci per liter.

The heavy water is not exposed to a significant fast neutron flux. Hence, the nuclides that are seen in the primary coolant (N-16, Na-24, Al-28) are not observed in quantity in the reflector.

c) Shield Coolant:

The shield coolant is not exposed to a significant neutron flux and its activity is usually less than detectable.

11.1.1.6 Solid Radioactive Sources

The principal solid sources of radioactivity are the MITR fuel elements and activated core components such as the absorber sections of the shim blades. Another category of solid sources are port plugs and the detection elements of nuclear instruments. Provisions exist for the safe storage of solid sources. Also, movement of these sources is governed by written procedures.

Fuel elements are typically in-core or in storage awaiting reuse for a total period of three to five years. On removal, after sixty days of cooling, dose rates are estimated at about 10,000 rem/hour at 1 meter unshielded. Spent elements are moved from the core tank to the fuel storage pool by means of a shielded transfer cask.

Reactor components such as shim blades, dummy elements, and port plugs are stored in various locations depending on the activities encountered. Port plugs, for example, may have activities ranging from less than 0.1 mCi to several curies comprised principally of ^{60}Co. Higher activity plugs are stored in a shielded wall storage position designed for this purpose. Port plugs are typically re-usable depending on the particular experimental needs and hence are not classified as waste. Shim blades, once depleted, are transferred to the spent fuel pool by means of a shielded cask and are stored within the fuel storage pool for decay until acceptable for shipment as waste. These shim blades can have activities (principally ^{60}Co, ^{58}Co, ^{59}Fe) on the order of 50 - 60 Ci of initial activity, depending on the operating history. Neutron detectors such as fission chambers and ion chambers have dose rates on removal from service that may range from less than 1 mrem/hour on near contact to several rem/hour at 1 meter. Storage locations may include the vertical dry storage locations on the reactor top area, wall storage if the dose rates require the item to be shielded, or other designated storage locations.

11.1.2 Radiation Protection Program

The Radiation Protection Program for the MITR is implemented by the Reactor Radiation Protection Office, which is part of MIT's Environmental Medical Service. The program is established pursuant to 10 CFR 20, Subpart B - Radiation Protection Programs, paragraph 20.1101 and the MITR Technical Specifications. Figure 12-2 is an organization chart for the MIT Reactor Radiation Protection Office (RRPO). As shown in the figure, and also by comparison with Figure 12-1 which depicts the organization for Reactor Operations, RRPO is administratively separate from Reactor Operations through the Vice-President level. The reactor radiation protection program falls under the general supervision of the Institute Radiation Protection Officer. Individuals within the RRPO who have specific responsibilities are as follows:

a) Reactor Radiation Protection Officer:

The Reactor Radiation Protection Officer is responsible for radiation protection at the reactor and advises the Director of Operations in all matters pertaining to radiation protection. The responsibilities include calibrations, surveys, effluent monitoring, experiment review, and the personnel monitoring program. Qualifications for the Reactor Radiation Protection Officer shall be as specified in Section 12.1.4(c) of this report.

b) Assistant/Associate Reactor Radiation Protection Officer:

The Assistant (or Associate) Reactor Radiation Protection Officer reports to the Reactor Radiation Protection Officer and assists in implementing the program. The Assistant or Associate Reactor Radiation Protection Officer shall serve as the designated alternate for the Reactor Radiation Protection Officer.

c) Reactor Radiation Protection Technicians:

Reactor Radiation Protection Technicians report directly to either the Reactor Radiation Protection Officer or the Assistant (or Associate) Radiation Protection Officer. Reactor Radiation Protection Technicians are responsible for performing day-to-day radiological surveillance requirements.

The Institute Radiation Protection Officer is a member of both the MIT Committee on Reactor Safeguards and the MIT Radiation Protection Committee. The latter establishes institute-wide radiation safety practices.

The MITR technical specifications state the authority of the Reactor Radiation Protection Officer to interdict or terminate activities that may compromise safety. The technical specifications also include a management commitment to an effective ALARA program.

11.1.2.1 Plans and Procedures

Plans and procedures for implementation of the radiation protection program and/or its component elements may be written either by the RRPO or the Operations Staff. The division of responsibilities is as follows:

a) Reactor Operations Staff:

The reactor operations staff initiates all written procedures for the major facility plans (emergency, security, operating, requalification, etc.) and for operation of the reactor (checklists, abnormal operating procedures, administrative procedures, tests and calibrations, and maintenance). These procedures are subject to the review process described in Section 12.3 of this report. That process includes review and approval by the Reactor Radiation Protection Officer whenever radiation protection issues are involved. An ALARA evaluation is included in each of these reviews when radiation protection issues are involved.

b) RRPO Staff:

The RRPO staff initiates all written procedures that solely affect radiation protection activities. These include calibration procedures for health physics instruments and test procedures for radiation monitors. When reactor operations activities are involved, the Reactor Operations Staff reviews and approves these procedures. This review is performed pursuant to Section 12.3 of this report if the equipment involves reactor operation (effluent monitor calibrations, for example).

11.1.2.2 Radiation Safety Training

Training requirements for personnel who have access to the reactor restricted area is in accordance with 10 CFR 19 and 10 CFR 20 and MIT policy as directed/endorsed by the Committees on Reactor Safeguards and Radiation Protection. Training is required only for those individuals who, during the course of employment, are likely to receive, in a year, an occupational exposure in excess of 100 mrem (1 mSv).

The training that an individual requires for entry to the reactor restricted area depends on the scope of the work that the person will perform. General categories are as follows:

a) Escorted Access:

Individuals who will be under continuous escort require only awareness training.

b) Firefighters:

Fire fighting personnel who would respond to an emergency at the facility receive periodic training that consists of a lecture on the facility layout, its hazards, and radiation safety. This is followed by a facility tour.

c) Police:

MIT maintains its own large police force and therefore provides all required police services including emergency medical response. All officers receive annual training that consists of a lecture on the facility, its hazards, and radiation safety. This is followed by both specialized training on the use of portable detectors and a facility tour. In addition, the MIT Campus Police actively participate in both medical and emergency plan drills and exercises.

d) Experimenters:

All prospective users of the reactor facility are required to receive the training specified by 10 CFR 19 prior to their actual use of the facility. This training is conducted jointly by the Reactor Radiation Protection Office and Reactor Operations.

e) Reactor Operators:

The initial training is the same as that for experimenters. Operators then undergo the training described in Section 12.1.5.1 of this report. That program includes training in radiation protection topics such as the use of protective clothing, methods to maintain exposures in accordance with ALARA, use of portable health physics instruments, contamination control, etc.

f) Senior Experimenters:

Senior Experimenters receive additional radiation protection training similar to that provided to reactor operators.

11.1.2.3 Review and Audit

The MIT Reactor Radiation Protection Officer performs and documents several internal audits that are provided to the MIT Committee on Reactor Safeguards. These include a program review pursuant to 10 CFR 20.1101 and reports on the use of special nuclear material.

An independent audit of the reactor radiation protection program is done annually by the MIT Committee on Reactor Safeguards as described in Section 12.2.4 of this report.

11.1.2.4 Record Keeping

See Section 12.6.2 of this report.

11.1.3 ALARA Program

A formal ALARA program has been established for the facility pursuant to 10 CFR 20.1101. This program is jointly administered by the Reactor Operations and Reactor Radiation Protection Offices. The program managers from the former are the Superintendent of Operations and Maintenance and the Director of Reactor Operations. Those from the latter are the Assistant/Associate Reactor Radiation Protection Officer and the Reactor Radiation Protection Officer. The program addresses all aspects of facility operation with specific emphasis on the following:

a) Procedure/Equipment Changes:

All changes to procedures and/or equipment that have safety significance are subject to a safety evaluation before the change can be made. This process, which is described in Section 12.3.2 of this report, includes a written safety review. This review in turn includes an ALARA impact statement.

b) Maintenance Outages:

All maintenance outages are preplanned, and a written schedule for all major activities is prepared. These schedules are the joint effort of both Reactor Operations and Reactor Radiation Protection. One objective is to schedule work involving radiation so as to minimize exposure. For example, if work is to be done in the vicinity of the primary coolant heat exchangers, it might be possible to schedule it late in an outage so as to allow for the decay of Na-24 activity. Or, as an alternative, temporary shielding could be installed.

c) Planning Meetings:

Joint Operations/RRPO meetings are held frequently (usually weekly and always before major outages) to discuss planning, the conduct of major work items, and procedures. Dose minimization is addressed at these meetings.

d) ALARA Subcommittees:

Whenever appropriate, an ALARA Subcommittee may be appointed from the Operations/RRPO staffs to investigate a particular issue and to make recommendations.

e) Effluent Doses:

An explicit limit for effluent exposure is established pursuant to 10 CFR 20.1101(d) at 10 mrem per year effective dose equivalent (EDE).

The ALARA program is one of the items reviewed by the MIT Committee on Reactor Safeguards during its formal meetings.

11.1.4 Radiation Monitoring and Surveying

11.1.4.1 Monitoring and Surveying Equipment

The facility is equipped with a full complement of instruments for radiation monitoring. These are used to monitor both work places and other accessible areas as well as effluent pathways.

All the instrumentation is calibrated at a specified frequency in accordance with written procedures that have been reviewed and approved by the process described in Section 12.3.2 of this report. These procedures and methods are generally in accordance with manufacturer recommendations and applicable industry standards.

Radiation monitors that provide an engineered safety feature are described in Section 7.7.2.1 of this report.

11.1.4.2 Monitoring and Surveying Program

The MITR radiation monitoring and surveying program includes surveys for radiation (alpha, β-γ, and neutron), contamination, air samples, and effluents. Both radiation and contamination surveys are performed by reference to facility maps that depict the area to be surveyed. These survey maps denote specific locations that are always surveyed so as to provide a reference for the detection of trends. In addition, the survey would include other locations with the choice being a function of both the type of work in progress and its location within the building. The survey frequency depends on the reactor operating schedule and whether or not work involving radioactivity is to be performed. Surveys are performed of accessible areas at least weekly during any week that the reactor is operating or that radiological work is in progress. Additional surveys are normally done daily and are quite often extended to full surveys if appropriate. In addition, special surveys are scheduled for maintenance activities that involve radioactive materials and/or the potential for personnel exposures.

Instrumentation used for the conduct of surveys includes meters (GM, ion chambers, proportional counters, and scintillation detectors) for direct radiation exposure, meters (pancake GM and gas flow proportional counters) for fixed and removable contamination, and laboratory instrumentation (low level alpha/beta gas flow proportional counter and regular alpha/beta sealed tube proportional counter) for removable contamination.

The performance of the above noted surveys is one means to assist in providing personnel protection. A second is the use of the area radiation monitors as described in Section

7.7.1 of this report. The system employs units with readout and warning capability within the control room to alert operators of an abnormal condition. The detectors also alarm locally to warn individuals who might be working in the area of an abnormal condition. Dosimetry is also located in various areas throughout the facility to monitor integral exposures. This dosimetry is similar to that described in Section 11.1.5 of this report.

Surveys and monitoring of effluent pathways are performed to ensure compliance with applicable sections of the regulations, license conditions, and any administrative controls. In addition, sampling of process systems such as the primary, heavy water reflector, shield coolant, off-gas, cleanup, and experimental systems is performed to monitor component performance. Activities may involve monitoring of a cleanup system to evaluate ion column performance, sampling of the primary and core purge system to monitor fuel performance, and analysis of the heavy water system to determine tritium concentration. Routine sample analysis includes air sampling and liquid sampling for general area, effluent, and process streams. Analysis of the samples may include gross alpha-beta counting, gamma spectroscopy, and liquid scintillation depending on the type and the activity of the sample. Liquid samples analyzed for tritium generally require distillation of the sample to preclude other contaminants in the analysis. Other sample preparations may include evaporation, dilution, and transfer. Trending is performed on a routine basis and those results requiring regulatory compliance are compared to the appropriate limits on a routine basis.

Sampling stations are located in minimally accessed areas and are generally ventilated and may incorporate shielding to maintain exposures to operating personnel ALARA. Sampling and analysis procedures are established for each of the routine samples and a schedule for the surveillance is established within the procedure.

Air sampling includes fixed continuous air monitors, portable continuous air monitors, and grab-sampling systems for particulates and gases (noble gases, iodine, and tritium). Calibration of these systems is performed on a periodic basis in accordance with written procedures. Other air sampling methods, such as the use of breathing zone air samplers for intake assessment, the use of microprocessor-based continuous air samplers, and the use of portable bubblers for tritium collection are employed as needed. Bubblers are located in the reactor equipment room and in the base of the ventilation exhaust stack.

11.1.5 Radiation Exposure Control and Dosimetry

11.1.5.1 Personnel Dosimetry

The requirement for the monitoring of exposure to radiation and radioactive material is embodied within 10 CFR 20.1502. In particular, 10 CFR 20.1502 specifies the minimum monitoring requirements for adults, minors, and declared pregnant women for both external sources and internal exposures. An additional requirement for monitoring external exposures is established for individuals who enter a high or very high radiation area. Specifically, individual monitoring devices shall be supplied and required for use by these individuals. Monitoring devices for determination of compliance with dose limits specified within 10 CFR 20.1201 shall be by a NVLAP accredited process as required by and pursuant to the requirement of 10 CFR 20.1501.

Only a small fraction of the individuals who enter the restricted area of the MIT Research Reactor actually require the use of monitoring devices pursuant to 10 CFR 20.1502. Therefore, the use of monitoring devices for most individuals is at the discretion of the licensee. Nevertheless, almost everyone who enters the restricted area is provided with a monitoring device. The reporting of occupational exposures pursuant to the requirements 10 CFR 20.2106 is maintained only for those individuals who require monitoring in accordance with 10 CFR 20.1502.

The requirement described above for the monitoring of external exposures also applies to the monitoring of intakes. However, there has never been an instance during the operating history of the MITR in which such monitoring was necessary. Monitoring of intakes for workers is performed at the convenience of the licensee in part as a quality assurance check of facility and personnel performance. In the event that the monitoring of intakes should be required pursuant to

10 CFR 20.1502, then methods for assessing intakes for purposes of summation of external and internal doses would be established in accordance with the requirements set forth under 10 CFR 20.1202. The monitoring and the assessment of intakes could be achieved through the MITR's existing air sampling and bioassay programs. The air sampling program is described in Section 11.1.4 of this report. The bioassay program, as established, is maintained at the discretion of the licensee. It may include in-vivo or in-vitro bioassay, as appropriate, depending upon the radionuclides anticipated. The frequency of bioassays are nominally baseline, annual, and termination. Bioassays are performed in addition to the normal bioassay frequency. They are based on specific facility evolutions and would include cases of suspected significant intakes predicated on the results of air sampling or personnel contamination events.

The use of planned special exposures as permitted within 10 CFR 20.1206 is not envisioned for the MITR and hence no provision is made for such exposures. In the unlikely event that a planned special exposure is desirable based on facility needs, than a planned special exposure program will be established in accordance with the requirements of 10 CFR 20.1206.

At present, the dosimetric monitoring of occupational personnel exposures at the MITR is accomplished by the use of devices that are provided and processed by a vendor. The vendor-reported exposures are the doses of record. These are reviewed by the MITR Radiation Protection Office for compliance with internal administrative limits, regulatory limits, and for trending. In addition, comparisons are made of the vendor-reported doses with those obtained from self-reading pocket dosimeters.

Current vendor-provided dosimetry incorporates the use of whole body and extremity monitoring that is capable of measuring the dose, energy, and type of radiations of interest at the MITR. CR-39 track-etch dosimeters coupled with a TLD albedo-dosimeter (LiF pair) allow detection of an extended range of neutron energies as well as β-γ information. The current vendor processing frequency is quarterly.

In addition to the vendor-supplied dosimetry, personnel are issued pocket dosimeters. One purpose for this practice is, as noted above, to provide a comparison with the dose-of-record.

In addition, this practice allows for the trending of personnel exposures and provides an estimate of accrued exposure. All personnel who are assigned pocket dosimeters are trained in accordance with the limitations on use and the procedures involved. Common practice involves reading pocket dosimeters prior to entering and upon exiting radiation areas, with frequent reading in high radiation areas. Lost and off-scale dosimetry is governed by procedure. Users of pocket dosimetry are required to record exposure readings at a predetermined administrative level and monthly. The dosimeters are calibrated on a periodic frequency, usually every six months. Personnel who are allowed un-escorted access to the facility are assigned pocket dosimeters and signify their presence in the facility by use of an accountability board at the entrance to the restricted area. Members of the general public are allowed to be escorted for tours of the facility and are assigned pocket dosimeters of which logs are maintained.

11.1.5.2 Exposure Control

The control of radiation exposure at the MITR encompasses the entire reactor radiation protection program. Many of these controls are addressed in this SAR and include basic facility design (containment, biological shield), procedural documentation, training, entry control devices, personnel protective equipment, exposure limits, and record-keeping. Of particular interest is the control of high and very high radiation areas (10 CFR 20, Subpart G). These controls may include local and remote audible alarms that are coupled with visual surveillance (reactor top), controlled key access to locked high radiation areas (equipment room and secondary chemistry area's shielded cell), and control devices to prevent unauthorized access to very high radiation areas (medical therapy irradiation rooms). In addition, both the use of precautionary procedures (10 CFR 20, Subpart J) and the storage and control of licensed material (10 CFR 20, Subpart I) are employed to control exposures from external sources.

Examination of personnel dosimetry results over the past ten years shows an average of 10.1 person-rem as a total annual exposure for the entire registered group, which has averaged annually 183 personnel. Since fiscal year 1992, no one has exceeded the 1.00 - 1.25 rem dose group. Also, more than 60% of those who have access to the MITR have exposures in the non-measurable range. The Reactor Radiation Protection Office maintains files for information such as personnel exposures, facility radiation/contamination surveys, instrument calibration files, environmental monitoring results, waste disposal records, and others which are described in Section 12.6.2 of this report.

11.1.6 Contamination Control

The frequencies for the performance of contamination surveys are the same as those for radiation surveys (Section 11.1.4 of this report). The stated survey frequencies are the minimum required. The actual frequencies depend on the work that is in progress and are often daily or even several times per day.

Direct frisking/monitoring of personnel is required at designated frisking stations upon exiting the reactor building. Personal items that have been taken into the controlled area are considered to be integral to the person and are monitored accordingly.

Methodologies for access control, the avoidance of contamination spread, and the remedying of contaminated areas are implemented through a number of measures that include training, postings, boundaries, coverings, containment, isolation, notifications, labeling, locking, and decontamination. Once identified, contamination sources are secured and the area is surveyed for magnitude and extent. These areas are decontaminated as soon as practical. Confirmatory surveys are then performed as appropriate.

Occupational workers are required to be monitored as soon as practical after exiting a controlled surface contamination area. If contamination is identified that suggests an intake which requires monitoring or assessment in accordance with procedure, then an estimate of the intake is made.

Anti-contamination techniques include preplanning, pre-staging, mock-up training, covering/coating tools, inspection of protective clothing, protective clothing, shielding, the use of HEPA vacuum cleaners, negative ventilation, the use of fume hoods, posting, use of engineering controls, step-off pads, cleanable surfaces, and labeling. The use of protective clothing and its disposition is governed by procedure and/or directive. Examples of disposition may be reuse, relaundering, or disposal.

Equipment and components that are contaminated and which are to be taken outside of contaminated areas are bagged or covered and either marked or labeled. Contaminated material and equipment may be designated for reuse, decontaminated for controlled reuse, decontaminated for unrestricted use, or dispositioned as waste. The criteria for this disposition is dependent on the dose rate, contamination levels, and resources available.

The training program for staff and experimenters includes discussion of the risks of contamination and associated decontamination techniques.

11.1.7 Environmental Monitoring

The MITR environmental monitoring program is established to ensure that doses to members of the general public are within the prescribed limits, 100 mrem/year TEDE. This program includes the monitoring of both effluents to the environs and direct exposure pathways.

The Radiological Environmental Monitoring Program (REMP) for the MIT Research Reactor is divided into effluent monitoring and environmental surveillance. Effluent monitoring is concerned with measuring the types and amounts of radionuclides released from the facility. Environmental surveillance is concerned with measuring radioactivity and/or radiation in the environment. One of the primary objectives of a surveillance program is to demonstrate compliance with applicable regulations. However, in the conduct of surveillance within the environment, it must be recognized that the program can only provide estimates of exposures to the population (usually as an upper bound) because emissions are so small that facility-related radioactivity is frequently undetectable. Even when small amounts of radioactivity or radiation are detected, the data are frequently accompanied by a great deal of uncertainty. Such uncertainty may be related to measurement type, number of measurements, instrumentation used, methods used for analyses, and confounding factors such as variation in ambient background levels. As a consequence of the inherent limitations of the surveillance programs, compliance with regulatory radiation dose limits during routine operation, particularly for airborne emissions, is usually demonstrated by use of a combination of measured emissions and environmental transport models.

MIT exposure pathway scenarios are defined for both an effluent exposure pathway and a direct exposure pathway. The effluent exposure pathway may be further defined as a direct plume exposure pathway and an intake exposure pathway, typically along the downstream transport of the plume. The direct exposure pathway is assumed to be from any radioactivity contained within the restricted area and may be the result of direct shine (uncollided dose) from the facility or indirect shine (skyshine, scatter, or collided dose) from the facility.

Regulatory requirements for doses to the public are specified in 10 CFR 20.1301, "Dose Limits for Individual Members of the Public." The total effective dose to individual members of the public should not exceed 0.1 rem (1 mSv) in a year from licensed operation exclusive of background and the dose in any unrestricted area from external sources should not exceed 0.002 rem (0.02 mSv) in any one hour. In addition to these requirements, 10 CFR 20.1302, "Compliance with Dose Limits for Individual Members of the Public," provides specific requirements for the demonstration of compliance as follows: " The licensee shall make or cause to be made, as appropriate, surveys of radiation levels in unrestricted and controlled areas and radioactive materials in effluents released to unrestricted and controlled areas to demonstrate compliance with the dose limits for individual members of the public in 20.1301."

10 CFR 20.1302(b)(2) gives values for limiting the direct exposure pathway dose to 0.002 rem in any one hour and 0.05 rem per year.

Environmental surveillance for the direct exposure pathway may include direct and passive monitoring techniques. Direct monitoring methods may include measurements of ambient radiation levels and/or contamination levels. Passive monitoring usually implies integrating devices, such as thermo-luminescent dosimeters (TLDs).

The TLDs used for the restricted area boundary at present are comprised of Al_2O_3 phosphors in an environmentally sealed enclosure/holder arrangement with a detection sensitivity down to 0.1 mrem for the reporting period. The integrating period for the TLDs is nominally monthly. Transit, deploy, and other control TLDs are used to subtract storage, deploying, and shipping exposures from the total response for each location. TLD locations are chosen to indicate possible direct exposures.

Direct real time monitoring of the effluent exposure pathway is achieved by using environmental radiation monitors that are located at various quadrants within a quarter mile radius of the MITR. This system is designed to detect radio-noble gases discharged via the reactor stack. The system presently in use is based on 5.5 inch long, thin-walled G-M detectors coupled to ratemeters at a remote site. The nominal wall thickness of the G-M detectors is 30-40 mg/cm^2 and each unit has a nominal response of 7000 cpm per mrem/hour (gamma). Each system has an output which is relayed via a dedicated phone line to the Reactor Radiation Protection Office, where the signal is monitored and recorded. This system complements the effluent exposure pathway surveillance program. Other methods for providing direct exposure pathway assessment may be employed as appropriate and may be used in lieu of the presently described system. In addition, environmental thermo-luminescent dosimeters (TLDs) are placed at these locations to monitor integral doses on a quarterly basis. Both the real-time and the integral TLD exposure measurements indicate the EDE is well within regulatory requirements and is consistent with the results of the "CAP88-PC" computer code [11-1].

Routine environmental surveys are presently performed quarterly. The extent of these surveys is predicated on the operational conditions of the facility and facility evolutions. These surveys may include both scanning and static measurements.

11.2 Radioactive Waste Management

11.2.1 Radioactive Waste Management Program

11.2.1.1 Requirements

The requirements for waste disposal as it applies to the MITR are specified within 10 CFR 20 Subpart K, Appendix G to 10 CFR 20, and by inference 10 CFR 61, 10 CFR 71, and applicable DOT regulations. Specifically, 10 CFR 20 Subpart K permits the disposal of licensed materials by methods specified within 19 CFR 20.2001(a). The methods employed by the MITR for the disposition of licensed material include transfer to an authorized recipient (10 CFR 20.2001(a)(1)), decay in storage (10 CFR 20.2001(a)(2)), release in effluents (10 CFR 20.2001(a)(3)), disposal by release into sanitary sewage (10 CFR 20.2001(a)(4) and 10 CFR 10.2003), and disposal of material designated as specific waste (10 CFR 20.2004).

11.2.1.2 Philosophy and Objectives

The objective of the MITR radioactive waste management program is to operate the facility in keeping with the ALARA concept. Thus, the program emphasizes both the avoidance of the creation of the radioactive waste and the minimization of any waste that is created. Minimization of wastes, particularly at the point of generation, not only reduces total volume of wastes and thereby costs, but also helps to enable the overall philosophy of maintaining all exposures as low as reasonably achievable. The ALARA philosophy regarding the disposition of wastes is extended to effluents, discharges to the sanitary sewer, exposures to the environment, and personnel exposures.

Training and awareness at all levels coupled with a strong management commitment and support are the most effective tools employed to achieve the goals and philosophy of waste minimization and ALARA.

11.2.1.3 Organizational Structure

Each administrative entity at MIT that has authorization to use radioactive materials is responsible for enforcing the objectives of the radioactive waste management program. For the MITR this means that the disposition of licensed materials is ultimately the responsibility of the Reactor Operations Group with guidance from the Reactor Radiation Protection Office. Procedures concerning the disposition, record keeping, and other requirements of the radioactive waste management program are prepared jointly by Reactor Operations and Reactor Radiation Protection. Control of waste generated by experimental groups at the MITR is the responsibility of the Senior Experimenter (i.e., Principal Investigator) for the group. These individuals, who are either faculty or research scientists/engineers, also receive guidance from the Reactor Radiation Protection Officer. The RRPO, the Senior Operation Staff, and the Director of the Nuclear Reactor Laboratory have the authority and the responsibility to interdict any practice that is not in keeping with program objectives.

For activities conducted at the reactor site, the MIT Committee on Reactor Safeguards (MITRSC) has review and audit responsibility. For activities conducted off-site, the MIT Committee on Radiation Protection (which oversees the MIT Byproduct License) has review and audit responsibility. The MITRSC is discussed in Section 12.2 of this report.

11.2.1.4 Waste Management Training

Waste management training is conducted for all personnel who work in or frequent the restricted area of the reactor facility. The training is commensurate with the job function of the individual. The extent of the training ranges from simple awareness concepts to extended instruction on the processing and disposition of waste. For example, training of students who may use a specific experimental facility that would not normally generate radiological waste would cover the disposition of non-radiological waste and the importance of segregating this waste at the point of generation. However, licensed operators and radiation protection personnel would be trained on all aspects of waste management.

11.2.1.5 Document Control Measures

Documents used for control, processing, and disposition of wastes are similar to all other controlled documents that are maintained for the MITR. These are described in Section 12.3 of this report. The minimization of radioactive waste is a goal of the facility management. Accordingly, the safety evaluations that are prepared for major experiments include a section that addresses waste issues. Similarly, if a reactor operating procedure can generate radioactive waste, then its safety evaluation will include a review of that generation.

Records of radioactive waste management are maintained jointly between the Reactor Operations Division of the MITR and the Reactor Radiation Protection Office. Most records are maintained for the life of the facility, particularly those that pertain to potential dose. Requirements for the retention of records are given in Section 12.6 of this report.

11.2.1.6 Reviews and Audits

Audits of the radioactive waste management program are conducted as part of the facility audit programs. These include the routine quarterly audit that is done by Reactor Operations of all reactor activities, the annual review of radiological controls, and the independent audit that is done by the MITRSC. All reports and audits, including the annual report to the NRC, are reviewed by the Reactor Safeguards Committee.

11.2.1.7 Record Keeping

Refer to Section 12.6 of this report. No wastes are stored for the life of the facility and none are buried on site.

11.2.2 Radioactive Waste Controls

11.2.2.1 Definition

One objective of the MITR waste management program is to minimize the volume of all radioactive waste. This means that an item that contains radioactivity is not defined as being waste until it has no further potential use. Alternatives to the declaration of a radioactive or contaminated item as waste include:

a) Decontamination.

b) Controlled reuse.

11.2.2.2 Gaseous Waste

The generation of gaseous or airborne waste at the MITR is described in Section 11.1.1.4 of this report. During normal operation of the MITR, airborne waste is released through the ventilation system's exhaust stack. During accident conditions, airborne waste could be released through containment building leakage and through the building pressure relief system. The ventilation system is described in Section 9.1 of this report. The pressure relief system is described in Section 6.5.4.2 of this report.

a) Normal Operation:

All ventilation exhaust streams are filtered. The filter banks in the building ventilation exhaust system consist of one bank of roughing filters and one bank of absolute (HEPA) filters. The roughing filters serve as a pre-filter for the absolute filters. The primary and secondary chemistry areas as well as the reactor floor hot cells are serviced by individual absolute filters before the air from those areas is merged with the building exhaust flow. The building ventilation exhaust system is also equipped with redundant sets of plenum and stack radiation monitors as described in Section 7.7 of this report. The plenum gas and particulate monitors are interlocked to stop ventilation and to isolate the containment building should any abnormal levels of radioactivity be detected. The stack monitor provides confirmation that no abnormal activity is being discharged.

b) Abnormal Operation:

The containment building would be isolated and, if an overpressure condition was approached, the pressure relief system would be placed on line. The effluent would be processed through roughing and charcoal filters for removal of radio-iodines.

The HEPA filters serve to reduce the release of particulate activity. These filters require periodic replacement and hence by themselves become a source of solid radioactive waste.

11.2.2.3 Dilution Factor

Use of straight-line Gaussian diffusion equations yields a dilution factor for the facility of $>1 \times 10^6$ at 900 m. This was obtained using the CAP88-PC code [11-1]. This large dilution factor is expected because of the very large stack height and the predominance of atmospheric Pasquill Stability Class D (74%). Calculations made in conjunction with the MITR-II startup allowed for the effect of nearby buildings. Those calculations gave a dilution factor of at least 50,000. Accordingly, a factor of 50,000 is selected. An effective dose scaling factor of 1,200 is applied for radioactive iodines and particulates with half-lives greater than eight days to account for differences in dose pathways and dose conversion ratios. This factor is determined by estimating the dose from all pathways for Iodine-131 compared to noble gases for a unit release, based upon values generated using the CAP88-PC code. This factor assures that, if the MITR stack releases of iodines and particulates with half-lives greater than eight days are kept within the 10 CFR 20 limits at the nearest point of public occupancy, the potential for radiation doses after dilution will be a small fraction of the 10 CFR 20 limits.

11.2.2.4 Liquid Wastes from the Reactor Building

Liquid wastes result from the sampling of coolants, decontamination activities, routine cleaning of the building such as the washing of floors, the use of sinks for personal hygiene, and air conditioner condensate. Any leakage from a coolant system would also produce liquid waste. There are no floor drains in the facility. Hence, any spills are cleaned up locally. Liquid wastes are usually not generated by experiments. If such wastes are generated, they could be directed to the liquid waste system or collected locally. The latter option would be used if high specific activities were involved. Disposition of these wastes may involve ion exchange, decay in storage, solidification, or transfer to an appropriate waste management broker. MITR liquid waste is only mildly radioactive and does not require any shielding.

All collection points for liquid waste (such as sinks) route the effluent to a sump that is located in the equipment room. The sump is equipped with a level-controlled pump that will permit the transfer of collected liquids to one of two 1000-gallon waste storage tanks once the liquid level in the sump reaches a predetermined level. Transfer of liquids from the sump to the tanks is monitored by an in-line radiation monitoring system that incorporates an interlock which, upon signal, will de-energize the pump thereby stopping the transfer and will cause an alarm that notifies the console operator of the condition. The radiation monitor's alarm is preset so that if both tanks were to be full of water whose activity was slightly below the alarm set point during the pumping process, radiation levels at the site perimeter would be below those permitted.

The two 1,000-gallon waste tanks are located outside the reactor containment building in a normally locked, above-ground enclosure that is located in the restricted area. The enclosure is heated and is provided with leak detection capability. The tanks are supported by a concrete pad within the enclosure. A two-inch lip is provided which is designed to contain the volume of the on-line tank in the event of a system leak. In the event of a leak, an alarm will indicate in the reactor control room.

Only one tank is aligned to the sump at a time. Once filled, it is staged for processing while the other tank is placed in service. In the event that one of the tanks overfills, a common vent line permits the overfill to be discharged to the standby tank. Each tank is also provided with float switches for a high level alarm. If both high level alarms are activated, the sump pump is de-energized. In addition, the waste storage tanks are closed systems that share a common vent that connects to the stack ventilation system. This vent, which has an in-line HEPA type filter, is monitored by the stack radiation monitoring system. In the event that all alarms and interlocks fail, overflow of the tanks would be directed to the stack base and not to the local surroundings.

A sewer discharge pump that is located in the restricted area allows liquid waste to be circulated within any individual tank, between the tanks, or to the sanitary sewer system. Protocols for the discharge of liquid waste from the tanks require that a sample be obtained for radiological analysis. If the liquid is within permissible limits, a discharge permit is generated. The discharge pathway is from the tank to be discharged, through a filtration system, through a radiation monitor for continuous monitoring, and then to the sanitary sewer. The final valve in the discharge path is normally locked to prevent inadvertent discharge. It is only unlocked and aligned in accordance with a written approved procedure and the discharge permit. If the radiation monitoring system setting is exceeded during the discharge process, it will de-energize the pump and cease any discharge.

Radiological analysis requires that the sample obtained from the waste discharge be representative of the discharge contents. Representative sampling is achieved by recirculating the tank volume using the sewer discharge pump and drawing a sample downstream of the filter system after first purging any dead leg volume. Typical radiological analyses for waste tank discharges include gross alpha-beta, tritium, and isotopic content in accordance with written and approved procedures. Analytical equipment used for these analyses is calibrated in accordance with written and approved procedures.

Ancillary wastes associated with any liquid system are processed for disposal as solid waste in accordance with 10 CFR 20 and 10 CFR 61. These wastes may include filters and resins from ion columns. Resins are dewatered and analyzed for content to meet the criteria for disposal in accordance with 10 CFR 61. Dose rates and activities from the various ancillary wastes will depend on the system. All systems that produce ancillary wastes that are capable of generating significant dose rates are maintained in a shielded configuration until they are processed as waste. This permits the minimization of dose and the decay of the short-lived high activity constituents prior to handling. Spare ion columns for both the primary system and the heavy water system are available. The processing or changing out of ion columns is performed with the spare unit to permit decay and thereby reduce the exposure to the personnel who perform these maintenance activities. Primary ion columns are changed approximately once a calendar quarter. The off-line ion column is then allowed to decay so as to eliminate Na-24 and thus reduce dose to maintenance and operation personnel during the re-pack phase of ion column replacement.

11.2.2.5 Secondary System Discharge

Secondary coolant is discharged to the sanitary sewer from the cooling tower in order to limit the buildup of dissolved solids. This discharge is not radioactive. However, its volume is measured and the secondary coolant is monitored to ensure that there is no leakage from a primary or D_2O heat exchanger to the secondary. A description of the radiation monitoring system and the action to be taken if heat exchanger leakage does occur is given in Sections 5.5.1.5 and 7.7.2.2 of this report.

In addition to the use of on-line monitors to detect radioactivity in the secondary, a sample is taken daily and analyzed for tritium on any day that secondary coolant is flowing through a D_2O heat exchanger.

11.2.2.6 Other Liquid Waste

An additional waste stream that is not included within the description of the reactor itself would be laboratories that support the mission of the MIT Nuclear Reactor Laboratory. These laboratories include the engineering laboratory, the reactor radiation protection office radio-chemistry laboratory, and the isotope preparation laboratory. All liquid wastes generated within these laboratories are sampled and analyzed for radiological content prior to disposal. Disposal may include collection in containers for discharge to approved locations within the reactor building, processing through a portable cleanup system (filter and ion column), or discharge to approved laboratory sinks. Sinks within laboratories that are used for disposal of radioactive liquids are both approved and designated by the Reactor Radiation Protection Office. All discharges to these sinks are logged with respect to concentration and activity and are included in reports of discharges to the sanitary sewer. All discharges from approved sinks are in accordance with 10 CFR 20 and license conditions.

11.2.2.7 Tritium Discharge Limit

Liquid waste is discharged to the municipal sanitary sewer systems from two waste storage tanks and from the cooling tower basin. Radioactive nuclide concentration limits set on the monitoring and sampling systems are such that conformity with the limitations specified in 10 CFR 20 is assured, with the possible exception of tritium. This exception could occur if the average concentrations released in both water and air are well below the permissible limits, but the total amount of tritium released to the sewer in a year exceeds the five curies permitted by 10 CFR 20 because of leakage of reflector D_2O into the secondary coolant.

The average discharge of water from the secondary coolant system of the reactor to the sanitary sewer system is approximately 13,000 gallons per operating day. This is diluted by an approximately equal volume of water from the Nuclear Engineering Building. The sewerage from the site enters the Cambridge sewer system where it is further diluted by discharge by the rest of MIT (at about 1 MGD) and by an unmeasured amount of storm drainage. The Cambridge sewerage enters a Metropolitan District Commission trunk sewer. The estimated discharge rate from this line, based upon the permitted discharge rate, is 436 MGD [11-2].

Thus, the reactor effluent is diluted by a factor of about $4.3 \times 10^8 / 1.3 \times 10^4 = 30{,}000$ at the ultimate point of discharge from the sewer system. The tritium concentration limit of 1 μCi/liter in the cooling tower water assures concentrations at the point of discharge from the sewer system will be well below the limit of 10 CFR 20.

11.2.2.8 Solid Wastes

Solid radioactive wastes have been traditionally classified with respect to their physical characteristics as either "wet" or "dry." Wet wastes include spent ion exchange resins and filters as described in Section 11.2.2.4 of this report. Dry wastes generally consist of ventilation air filters, contaminated cloth, paper, metals, wood, plastic, and other items used in the conduct of normal facility operation. These are referred to as dry active waste (DAW). Other operational materials requiring disposition may include replaced reactor components, experimental apparatus, and generated wastes during facility modification.

The management of solid wastes can be categorized into four basic processes: collection, pretreatment, solidification, and packaging, which includes container handling, storage, and disposition. Waste minimization processes occur during all phases of the waste stream. Volume reduction methodologies for this facility include segregation/sorting, decontamination, and compaction. The process of segregation/separation involves the separation of non-compactible materials, compactible materials, metals, incinerable items, and clean waste. The process of segregating clean waste from the bulk waste may include the disassembly of components so that non-contaminated or non-activated parts are separate from those that are contaminated or activated. Solidification methodologies for higher specific activity wastes incorporates concrete as the solidification agent. Waste brokers are employed periodically to process certain wastes to achieve greater minimization of volume. Waste brokers are also used to process items that require special handling. Such items may include, for example, large components. Waste containers for the packaging of wastes may include 55 gallon drums, LSA type boxes, or the item itself if it can qualify as a package for LSA class wastes.

Exposure control is maintained at all points of the waste stream process so as to minimize dose to operating personnel and to the public. In particular, decay time is one of the most effective dose control tools available. By permitting the largest source term, Na-24, to decay prior to processing components or waste systems, the effective dose equivalent is reduced. Materials can be stored in shielded areas or in designated locations that are outside the normal working environment. All materials stored in these areas are appropriately marked and posted with respect to the anticipated hazards and radiological controls for entry or handling as appropriate.

11.2.2.9 Waste Minimization

The minimization of wastes helps ensure that the ALARA concept is maintained for operating personnel, other personnel, and the environment. Minimization also mitigates the quantities of material required for disposal associated with eventual facility decommissioning. The MITR program recognizes that the most effective method for the minimization of wastes is the reduction of source terms. Post-generation methods such as decontamination, segregation/sorting, and surveying are also employed.

Waste minimization techniques are categorized as administrative methods, dry-waste minimization techniques, and wet-waste minimization techniques. Administrative methods include the restriction of clean materials brought into radiological control areas, house-keeping programs, minimization of disposable items in contamination control zones, reduction of the size of contamination control zones, minimization of access points into the radiological control areas, proper pre-job planning, identification of sources, training programs, the use of audits, the use of signs and memoranda, and the establishment of release limits. The minimization of dry wastes includes the segregation of contaminated trash drums, recycling or re-using materials, sorting, decontamination of tools and equipment, and the use of readily decontaminatable items or equipment. Wet waste minimization techniques include leak rate management programs, leak detection programs, and a leakage repair program.

11.2.3 Release of Radioactive Waste

The release of radioactive waste is described in Section 11.2.2 of this report. In summary, airborne releases are primarily through the ventilation system exhaust stack. Continuous air monitoring is coupled with sample analysis. Automatic actions for containment isolation based on signal input from the plenum effluent monitors ensure that no releases would exceed any established limits. The sample analysis verifies the isotopic distribution and concentrations of the releases as measured by the effluent air monitoring system. Liquid wastes, which are discharged to the sanitary sewer, use a discharge permit process which requires sampling and continuous monitoring during the discharge process. This ensures that the discharge is in accordance with the provisions set forth within 10 CFR 20. Solid wastes are transferred to designated burial sites and all wastes are prepared for transfer and disposal in accordance with 10 CFR 61.

Annual reports filed in accordance with the MITR Technical Specifications have demonstrated that all releases and waste generation are within limits and that ALARA provisions are met. The increase in reactor power from 5 MW to 6 MW is expected to result in a 20% increase for airborne releases. This level is still maintained well below allowable limits. Releases via the sanitary sewer and for transfer to a licensed facility for disposal should not increase proportional to the power level increase and are maintained within applicable regulatory requirements.

References

11-1 Parks, B., "Mathematical Models in CAP88-PC," U.S. Department of Energy, June 1997.

11-2 Massachusetts Water Resources Authority, " Monthly Compliance and Progress Report, " June 1999.

Chapter 12

Conduct of Operations

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Chapter 12

Conduct of Operations

12.1 Organization

12.1.1 Structure

Figure 12-1 shows the organizational structure of the Reactor Operations Group. This group is one entity within MIT's Nuclear Reactor Laboratory (NRL). The NRL is an interdepartmental laboratory, of which there are several at MIT, that is under the direction of the Vice-President for Research. This arrangement makes the MIT Research Reactor available to faculty and students from all of MIT's academic departments.

Overall supervision of the reactor project lies with the Administration of the Massachusetts Institute of Technology. This chain of administrative authority and responsibility includes successively the President, the Provost, the Vice-President for Research, the Director of the Nuclear Reactor Laboratory, and the Director of Reactor Operations.

Overall direction on matters of reactor safety rests with the MIT Reactor Safeguards Committee (MITRSC). Approval of the MITRSC is necessary for all new operating plans and policies and all significant modifications thereto which may involve questions of nuclear safety. The MITRSC is also responsible for auditing operation of the reactor. The Chairman of the MITRSC reports directly to the President of MIT.

Nuclear medicine, especially neutron capture therapy, is a major activity at the NRL. Three MIT committees have oversight responsibility for aspects of this program. One is the MITRSC. The other two are the Committee on the Use of Humans as Experimental Subjects (COUHES) and the Committee on Radiation Exposure to Human Subjects (COREHS). The Chairmen of both COUHES and COREHS also report directly to the President of MIT.

Figure 12-2 shows the organizational structure of the Radiation Protection Office. One unit of this group, the Reactor Radiation Protection Office, or RRPO, is physically located at the reactor site. The RRPO group works closely with Reactor Operations. However, the two groups

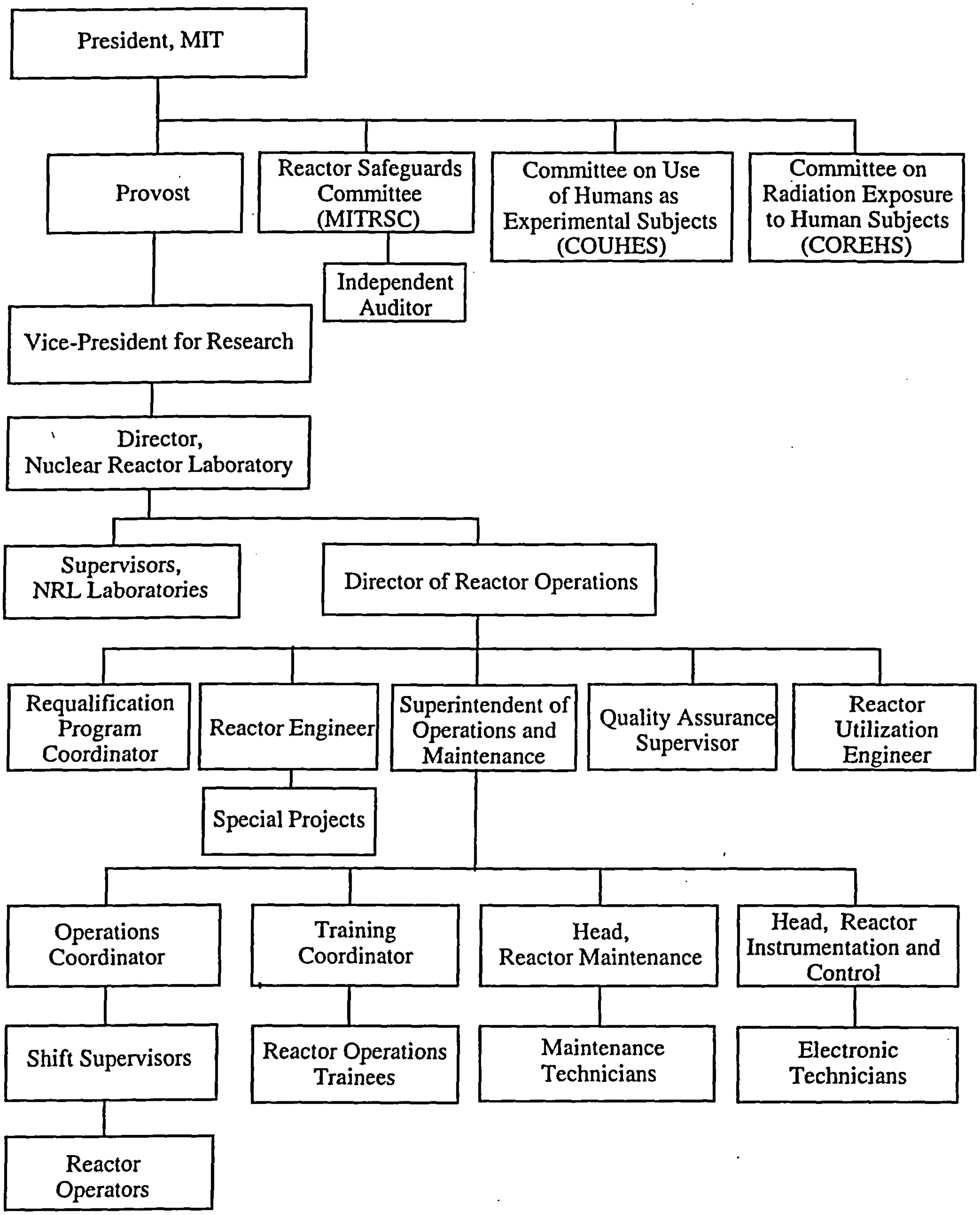

Figure 12-1. Organization Chart for MITR Operations

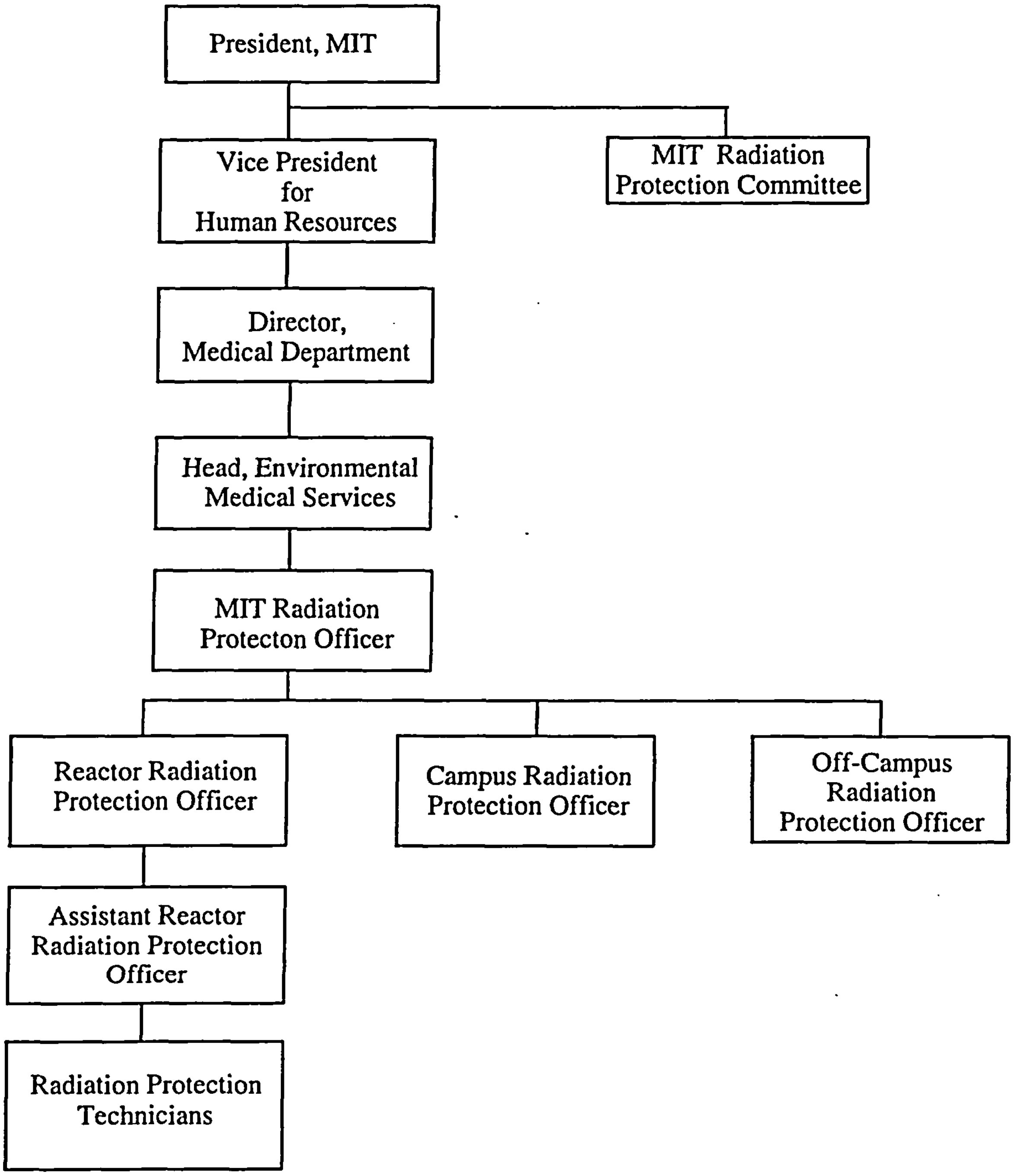

Figure 12-2. Organization Chart for MIT Reactor Radiation Protection Office

are administratively distinct with RRPO being part of the MIT Medical Department. This separation precludes the possibility of a conflict of interest between RRPO and Reactor Operations. The Radiation Protection Group is organized structurally in the same way as Reactor Operations. Overall supervision lies with MIT Medical and specifically, with the Environmental Medical Office. Matters of radiation safety require the approval of the MIT Radiation Protection Committee.

12.1.2 Responsibility

12.1.2.1 Reactor Operations

The following individuals or groups that appear in the organizational structure shown in Figure 12-1 have day-to-day responsibility for the safe operation of the facility. This includes protection of the health and safety of both the public and workers at the facility, as well as protection of the environment.

a) Reactor Safeguards Committee - The MIT Committee on Reactor Safeguards verifies that nuclear reactor operation is consistent with MIT policy, rules, approved operating procedures, and license provisions; unusual events are investigated promptly and corrected in a manner which reduces the probability of recurrence of such events; trends are detected which may not be apparent to day-to-day observers; and periodic audits of the operation of the reactor have been properly conducted. The Committee also reviews and approves all new experiments involving significant changes in procedure before the experiment is conducted.

b) Vice-President for Research - The Vice-President for Research controls the budget for the MIT Nuclear Reactor Laboratory. This individual is a member of the MITRSC and receives input from the MIT Radiation Protection Committee.

c) Director, Nuclear Reactor Laboratory - The NRL Director is responsible for all aspects of the NRL with special emphasis on research activities. This individual is a member of the MITRSC.

d) Director of Reactor Operations - The Director of Reactor Operations or DRO is responsible for overall direction of reactor operations, for operation of the reactor facility, and for control of the reactor fuel. This individual is the designated correspondent for the Reactor Operating License. The DRO reviews and approves all changes to procedures, equipment, and experiments. The DRO is a member of the MITRSC.

e) Superintendent of Operations and Maintenance - The Superintendent is responsible to the Director of Reactor Operations for the operation, maintenance, and refueling of the reactor and fission converter. The Superintendent has the immediate responsibility for scheduling reactor use and maintenance, for coordinating the performance of experiments and sample irradiations, for the proper packaging and shipment of radioactive material (except fuel), for the conduct of all instrument tests and calculations, and for operator training. The Superintendent is assisted by the Operations and Training Coordinators as well as by the Heads of the Reactor Maintenance and Instrumentation/Control sections.

f) Operations Coordinator - The Operations Coordinator is a shift supervisor who, in addition to directing his or her own shift, ensures that coordination exists with other shifts so that projects and or irradiations that require several shifts to complete are done expeditiously. In addition, this individual schedules sample irradiations and coordinates facility tours.

g) Training Coordinator - This individual oversees the training program for newly hired personnel at both the operator and senior operator level.

h) Head, Reactor Maintenance: This individual coordinates major maintenance activities.

i) Head, Reactor Instrumentation and Control: This individual coordinates the acquisition and installation of new instruments and control devices.

j) Shift Supervisors - Each shift is under the direction of a Shift Supervisor who has the responsibility for proper and safe operation of the reactor and its associated experiments during the shift. Other specific responsibilities include supervision of maintenance, direction of tests and calibrations, scheduling of sample irradiations, and supervision of fuel transfers and control element replacements.

k) Reactor Operators - The prime responsibility of a reactor operator is to assure safe operation of the reactor under normal conditions and to be able to take proper and timely action in the event of an abnormal situation.

l) Reactor Engineer - The Reactor Engineer is responsible for the performance of fuel management calculations and activities related to the procurement and shipping of fuel. This individual is also assigned as the coordinator for special projects such as the fission converter facility.

m) Reactor Utilization Engineer - The Reactor Utilization Engineer is primarily responsible for utilization of the reactor.

n) Quality Assurance Supervisor - The Quality Assurance Supervisor monitors all safety review and quality assurance records, ensures the proper distribution of safety reviews, and assists the DRO with internal audits of the facility.

o) Requalification Program Coordinator - The Requalification Program Coordinator is responsible for the implementation of the requalification program.

The Reactor Engineer and Reactor Utilization Engineer normally have the same background qualifications as the Superintendent and may be assigned by the Director of Reactor Operations to assist the Superintendent when necessary.

12.1.2.2 Radiation Protection

The following individuals that appear in the organization chart shown in Figure 12-2 have day-to-day responsibility for the radiological aspects of the facility's operation.

a) Reactor Radiation Protection Officer - The Reactor Radiation Protection Officer is responsible for radiation protection at the reactor and advises the Director of Reactor Operations in all matters pertaining to radiation protection. The responsibilities include calibrations, surveys, effluent monitoring, experiment reviews, and the personnel monitoring program. An Assistant Reactor Radiation Protection Officer and Radiation Protection Technicians assist in these duties.

b) MIT Radiation Protection Officer - The MIT Radiation Protection Officer oversees all uses of radiation at MIT, both on and off-campus. This individual is a member of the MITRSC and advises the NRL Director on issues of radiation safety.

In addition, there is a Radiation Protection Committee that establishes policies and provides training for activities that are not conducted at the reactor site.

12.1.3 Staffing

The minimum staffing of the facility is a function of the condition of the reactor. Two conditions are formally recognized in the MITR Technical Specifications. These are "shutdown" and "secured." The term "shutdown" is defined as that condition where all control devices are fully inserted or a reactivity condition exists that is equivalent to one where all control devices are fully inserted. The term "secured" is defined as that condition where there is no fuel in the reactor core or where the following conditions are satisfied: (1) the reactor is shut down, (2) the console key switch is off and the key is in proper custody, and (3) no work is in progress within the main core tank and/or the fission converter tank involving fuel or experiments or maintenance of the core structure, installed control blades, or installed control blade drives when not visibly decoupled from the control blade. If the reactor is not in one of these conditions, then it is assumed to be operating even though such may not be the case. The staffing requirements are:

a) When the reactor is not shut down, the minimum crew complement for a shift shall be two licensed operators including at least one licensed senior operator, one of whom shall be in the control room. In addition, the Reactor Radiation Protection Officer or a designated alternate shall be onsite or on call.

b) Whenever the reactor is not secured, two persons must be onsite, one of whom shall be a licensed senior operator. An operator or senior operator shall be present in the control room. In addition, the MITR Radiation Protection Officer or a designated alternate shall be onsite or on call.

Staffing at the MIT Research Reactor meets, at a minimum, the requirements of 10 CFR 50.54 (i), (j), (k), (l), and (m)(1). In particular,

a) Except as allowed in 10 CFR 55.13 for training, the reactor controls may be manipulated only by licensed operators or senior operators.

b) All personnel with active licenses must adhere to the requirements of the MITR operator requalification program.

c) Operation of apparatus (other than the reactor controls) that may affect reactivity may be done only with the permission of a licensed operator or senior operator. (Note: An example of such a manipulation is the cycling of shutters for the medical irradiation facilities.)

d) A licensed operator or senior operator is required to be present in the reactor control room whenever the reactor is not shut down.

e) Supervisory personnel who are responsible for directing the licensed activities of licensed operators are required to hold active senior operator licenses.

f) A licensed senior operator is required to be at the facility whenever it is not shut down and to be present in the control room during all reactor startups. This includes recovery from unplanned or unscheduled shutdowns or from significant reductions in power.

12.1.4 Selection of Personnel

Minimum educational and/or experience requirements are established for those individuals who have line responsibility and/or authority for the safe operation of the facility.

a) Director of Reactor Operations - The Director of Reactor Operations shall have a minimum of seven years of nuclear experience. The individual shall have a recognized baccalaureate or higher degree in an engineering or scientific field. Education or experience that is job-related may be substituted for a degree on a case-by-case basis. The degree may fulfill four years of the seven years of nuclear experience required on a one-for-one time basis. At least three years of experience shall be in a responsible position in reactor operations or a related field including at least one year's experience in reactor facility management or supervision. The Director of Reactor Operations shall hold a senior operator's license for the MIT Research Reactor, or have held such a license at the MIT Research Reactor.

b) Superintendent of Operations and Maintenance - The Superintendent shall have a minimum of five years of responsible reactor experience. A maximum of two years of experience may be fulfilled by academic or related technical training on a one-for-one basis. The Superintendent shall hold a senior operator's license for the MIT Research Reactor.

c) Reactor Radiation Protection Officer - The Reactor Radiation Protection Officer shall have a minimum of five years of experience in radiation protection including at least one year of experience at a nuclear reactor facility. A maximum of four years of the five years experience may be fulfilled by related technical or academic training.

d) Shift Supervisor - Shift supervisors shall have a minimum of a high school diploma or equivalent and three years of responsible reactor experience. A maximum of 1-1/2 years experience may be fulfilled by academic or related technical training on a one-for-one basis. The requirement for experience may be completely satisfied by an advanced degree in Nuclear Engineering. Shift supervisors shall hold an NRC senior operator license for the MIT Research Reactor.

e) Reactor Operator - Reactor operators shall have a high school diploma or equivalent. They shall hold an NRC reactor operator license for the MIT Research Reactor.

12.1.5 Training of Personnel

12.1.5.1 Initial Training of Operators and Senior Operators

The conduct of training for newly hired personnel at both the operator and senior operator level is the responsibility of the Training Coordinator. This individual reports directly to the Superintendent for Operations and Maintenance. In addition, this individual works closely with the Requalification Program Coordinator. All individuals who hold a senior reactor operator's license are expected to assist with the training effort, especially with the performance of oral checkouts and the conduct of practical factors training.

The endpoint of the MITR training program for newly hired personnel is a license exam that is conducted by the U.S. Nuclear Regulatory Commission. The administration of these exams by the NRC provides an independent check on the quality of the training itself.

Formal training programs exist for reactor and senior operators. The basis of each is a military-style qualification card. The benefit of this approach is that the process of qualification is modularized. Hence, the material can be learned in a rational sequence and the progress of the trainees can be readily monitored. Reactor systems, reactor theory, and reactor procedures are broken down into modules with each module requiring a few days to at most a week of study. The trainee is expected to read all relevant material, to trace any systems, and to discuss the material with licensed personnel. The trainee is then given an oral examination by a senior operator. If that exam is satisfactory, the trainee takes a written test on the module. Periodic oral reviews are given to ensure that the trainee is integrating modules. In addition to this academic-type study, each qualification card requires the candidate to perform specific practical factors to the satisfaction of a senior operator. For example, operator candidates are required to perform startup checks of the I&C equipment and to start up the reactor. Upon completion of the entire card, the candidate is given a comprehensive written exam and a walkthrough by a member of the Senior Review Board.

Individuals may qualify for an "instant" senior operator license if they have the necessary prior experience. In this case, they are required to complete both the operator and senior operator qualification cards except for those portions of the former that are duplicated in the latter.

The above qualification programs meet or exceed the requirements of 10 CFR 55.

12.1.5.2 Requalification Training

Refer to Section 12.10 of this report.

12.1.5.3 Training Required by 10 CFR 19

All prospective users of the reactor facility are required to receive the training specified by 10 CFR 19 prior to their actual use of the facility. This training is conducted jointly by the Reactor Radiation Protection Office and Reactor Operations. The content of this training program is established by the MIT Radiation Protection Committee. Applicants are provided with reading material and references on the fundamentals of radiation protection, radiation limits, and the handling of radioactive material. Candidates then study and/or discuss the material. They then receive an oral checkout from the Reactor Radiation Protection Officer. If that checkout is satisfactory, candidates are then given both a facility tour and instruction on evacuation and emergency response. All training is documented.

12.1.5.4 Specialized Training

Non-licensed individuals may wish to utilize certain of the reactor's experimental facilities. Formal qualification cards have been prepared to train these individuals. These cards are structured in the same way as the operator/senior operator qualification cards and the training approach is the same. Cards currently exist for the medical irradiation facilities and beam ports.

12.1.5.5 Senior Review Board

The Senior Review Board (SRB) consists of the Training Coordinator, the Requalification Program Coordinator, the Superintendent of Operations and Maintenance, and the Director of Reactor Operations. The SRB is responsible for the overall quality of the MITR initial training and requalification programs. In particular, the SRB does the following:

a) Monitors the quality of the candidate training program.

b) Monitors trainee progress and makes the decision on when to request an NRC-administered exam for each trainee.

c) Reviews the results of NRC-administered exams, and if retesting is needed, approves the content of the retraining effort.

d) Monitors the quality of the requalification program lecture series.

e) Reviews records of on-the-job training.

f) Reviews the scenario for the emergency plan exercise and the report of that exercise.

g) Reviews the content of all written exams including those given for requalification.

h) Ensures exams are graded in a timely manner.

i) Reviews the results of the requalification program (both written and walk-through exams) and, if upgrading is needed, approves the content of the upgrade effort.

In addition, at the end of each requalification cycle, the SRB reviews all aspects of each licensed individual's performance to determine his or her capability to continue to discharge duties in a safe and competent manner.

12.1.6 Radiation Safety

The organization of the radiation safety program is described in Section 12.1.1 of this report with additional information given in Chapter 11 of this report.

12.2 Review and Audit Activities - MIT Reactor Safeguards Committee

The committee responsible for review and audit of the reactor facility is the MIT Committee on Reactor Safeguards or MITRSC. The role of the MITRSC is summarized in Section 12.1.1 of this report. The MITRSC is vested by the President of MIT with approval authority for design, operation, and use of the MIT Research Reactor. The MITRSC reports directly to the President of MIT and through the authority of the President's Office is charged to report its findings to the NRL Director, Director of Reactor Operations, and the MIT Vice-President for Research.

12.2.1 Composition and Qualifications

The Committee shall be composed of a minimum of nine persons with not more than one-third of the total membership chosen from the reactor staff organization and a minimum of three members from outside MIT. All members and the Chairman shall be selected by the President of MIT. At least four voting members including participating alternates shall have a minimum of a Bachelor's Degree in Engineering or the Physical Sciences and have a minimum of three years of professional level experience in nuclear services, nuclear plant operation, or nuclear engineering, and the necessary overall nuclear background to determine when to contact consultants for analyses beyond the scope of the Committee's expertise. Ex-officio members of the MITRSC shall include the MIT Radiation Protection Officer and the MIT Safety Officer.

Members are to be chosen so that the MITRSC collectively has the capability required to review matters in the areas of reactor operations including criticality safety, nuclear engineering, chemistry and radiochemistry, materials, instrumentation and control, radiation protection and health physics, mechanical and electrical engineering, environmental considerations, radiobiology and nuclear medicine, and industrial safety.

12.2.1.1 Quorum

A quorum shall consist of no less than a majority of the Committee's voting membership. In addition, either the Chairman or his designated alternate shall be present. Finally, no more than a minority of the quorum shall have line responsibility for reactor operations.

12.2.1.2 Subcommittee

The full MITRSC may, after discussion of a particular topic, designate a subcommittee to conduct further review prior to final review by the full Committee. The MITRSC may also choose to delegate approval authority to a subcommittee. The membership of a subcommittee, its chairman, its purpose, and its authority must be stated in the minutes of a meeting of the full MITRSC. These items must be reaffirmed at least annually. The quorum rule that applies to the membership of the MITRSC as a whole also applies to the membership of any subcommittee.

12.2.2 Charter and Rules

The MITRSC's Charter enumerates the subjects within the purview of the Committee, its responsibility and authority, its composition and membership qualifications, provisions for the use of subcommittees, authority for access to reactor records, reporting requirements, meeting frequency and mechanisms for convening meetings. The first four of these topics have already been described. As for the others:

a) Access to Reactor Records: The Committee has complete access to all records pertaining to reactor operation.

b) Reporting Requirements: Minutes are kept of all Committee (and subcommittee) meetings. These are distributed to all members of the Committee, to the MIT Administration, and to the Reactor Operations/RRPO Staff.

c) Meeting Frequency: Meetings shall be held at least annually. Meetings are normally scheduled by the Director of Reactor Operations. However, any committee member or any member of the Reactor Operations or Reactor Radiation Protection Office staff may bring a matter to the Chairman's attention with a request for a meeting.

12.2.3 Review Function

The Committee reviews all aspects of reactor operation. Items normally reviewed include minutes of the previous meeting, minutes of subcommittee activity since the last meeting, subcommittee authority, proposed facility changes including new experiment designs, proposed changes to procedures especially those to the emergency and security plans and to the training program, inspection reports from the U.S. Nuclear Regulatory Commission, internal audits of reactor operations and the health physics program, records of reportable occurrences, summaries of emergency plan exercises, and reports to the U.S. Nuclear Regulatory Commission (Annual Report). The Committee tracks all open items from previous meetings. Specific aspects of the MITRSC's review function include the following:

a) Determinations that changes in equipment, systems, tests, experiments, or procedures described in the annual report do or do not involve an unreviewed safety question (10 CFR 50.59).

b) All new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment, or systems having safety significance.

c) All new experiments or classes of experiments that could affect reactivity or result in the release of radioactivity.

d) Proposed changes in technical specifications or license.

e) Violations of technical specifications or license as well as violations of internal procedures or instructions having safety significance.

f) Operating abnormalities having safety significance.

g) Reportable occurrences.

h) Audit reports.

12.2.4 Audit Function

Many of the procedural documents that govern reactor operation require periodic audits by the Reactor Operations or the Radiation Protection Office staff. These include but are not limited to audits of reactor operation and the use of special nuclear material on a quarterly basis, an annual audit of the quality assurance program including the shipment of radioactive materials, annual reviews of the emergency plan and the training program, and a biannual review of the security program. The latter is done jointly with the MIT Campus Police. These internal audits are provided to either the MITRSC for review or to the independent auditor who is appointed by the MITRSC.

The MITRSC conducts its own independent audit. It may do this by designating certain members (ones not having line responsibility for the reactor) to conduct the audit or by hiring an outside consultant who has the requisite expertise. The scope of this audit includes but is not limited to all aspects of facility operations (console log, checklists, test and calibration records, equipment repair records, system tagouts, refuelings, reportable occurrences); the operator training program including the requalification program; actions taken to correct facility deficiencies; the emergency plan; the physical security plan; and the radiation protection program including surveys, effluent records, and personnel exposure summaries. The minimum frequency for this audit is annual.

12.3 Procedures

12.3.1 Scope of Procedures

Written approved procedures govern all aspects of the reactor facility's operation and use. These procedures encompass, but are not limited to, the following areas:

a) Standard operating procedures including both reactor startup and shutdown and fuel handling.

b) Checklists for reactor operation including instrumentation and mechanical equipment.

c) Security plan procedures.

d) Emergency procedures.

e) Abnormal operating procedures which specify actions to be taken by both the reactor operator and the supervisor (senior operator) in the event of a reactor or experiment alarm.

f) Procedures for performing instrument tests and calibrations as well as tests required by the technical specifications.

g) Maintenance procedures.

h) Procedures for the insertion and removal of in-core experiments, for the use of pneumatic tubes and other irradiation facilities, for the use of the medical irradiation facilities, for the use of the reactor floor hot cells, and for the use of experimental equipment located within either the reactor containment building or the restricted area.

12.3.2 Approval Process

Procedures are prepared by either the Reactor Operations Staff or the Reactor Radiation Protection Office or both. If appropriate, the draft process will include a walk through in which the procedure that is to be performed is simulated. This is done both to be certain of including all required steps and to avoid implementations that are impractical. Once a draft has been completed, it is subjected to a formal review process. A written document, designated as a "safety review" is produced. This review includes:

a) A summary of the procedure with emphasis on its purpose,

b) A safety assessment of the procedure,

c) An ALARA review,

d) A determination if an unreviewed safety question is involved, and

e) A determination if the proposed procedure will decrease the effectiveness of the emergency, security, or requalification plans.

The safety review requires the signatures of two senior operators, the Reactor Radiation Protection Officer (if radiation is involved), and the Director of Reactor Operations. The next step in the approval process depends on the nature of the procedure. All proposed changes are divided into two groups. The first group consists of those changes that would alter the standard operating plans, the physical security plan, the requalification program, the emergency plan, or the technical specifications. In addition, procedures that are outside the envelope of those previously approved by the MITRSC are in this group, for example, procedures for the conduct of a new type of experiment. The second group consists of everything else. Procedures in the first group are scheduled for discussion and review by the MITRSC. They may be issued only after receiving MITRSC approval. Procedures in the second group may be issued after the safety review subject to the MITRSC's being subsequently informed of the change. The notification to the MITRSC is normally done in the summary of procedural changes contained in the Annual Report submitted to the U.S. Nuclear Regulatory Commission.

Procedural changes that would affect the basis of a technical specification or otherwise involve an unreviewed safety question require approval of the U.S. Nuclear Regulatory Commission prior to issuance. Any procedures in this category are submitted to that organization after first being reviewed by the MITRSC.

12.3.3 Procedure Change Method

Substantive changes to procedures are made using the process described above in Section 12.3.2 of this report. Temporary deviations to deal with special circumstances may also be necessary and those may require implementation before the complete review process can be performed. In that case, the following guidance applies.

a) Every attempt should be made to follow approved procedures as written. Only in cases where an obvious mistake has been made or an unsafe condition could arise should changes be made "in the field." As a general rule, changes should not be made simply for convenience. In no case should changes be made which alter the intent of the original procedure.

b) Temporary change to procedures that do NOT require MITRSC review may be made by approval of two members of the reactor staff, at least one of whom shall hold a senior operator license. Such changes must be initialed by these personnel on the procedure copy used.

c) Temporary changes to procedures that require MITRSC approval may be made after review by two senior operators and the Director of Reactor Operations (DRO). The DRO will ensure a timely review by the MITRSC. Such changes must be initialed by these personnel on the procedure copy used.

d) Temporary changes to procedures that require NRC approval shall not be made except as permitted under 10 CFR 50.54 (x) and (y) which authorize a licensee to take reasonable action that departs from a license condition or a technical specification in an emergency when this action is immediately needed to protect the public health and safety and no action consistent with license conditions and technical specifications that can provide adequate or equivalent protection is immediately apparent. Such action shall be approved by a licensed senior operator prior to taking the action, and if time permits, it should be approved by the most senior licensed MITR staff individual on site or reachable by telephone. In the event of such a departure~~, note that~~ NRC must be notified, beforehand if possible (10 CFR 50.72(b)(i)(B)).

12.3.4 Equipment Changes

The review and approval process (Sections 12.3.1 and 12.3.2) shall apply to changes of equipment that can affect reactor safety.

12.4 Required Actions

As noted in Section 12.3.1 of this report, written procedures exist that are to be followed in the event of a departure from normal operation. Among these procedures are ones that address safety limit violations and other reportable events.

12.4.1 Safety Limit Violation

In the event of a safety limit violation, the MIT Research Reactor shall be shut down and the senior management of the facility notified. The latter includes the Superintendent, the Director of Reactor Operations, the Director of the Nuclear Reactor Laboratory, and the Chairman of the MITRSC. The event shall then be investigated and a safety limit violation report prepared. The report shall describe the following:

a) Applicable circumstances leading to the violation including, when known, the cause and contributing factors.

b) Effect of the violation upon reactor facility components, systems, or structures, and on the health and safety of personnel and the public.

c) Corrective action to be taken to prevent recurrence.

This report shall be reviewed by the MITRSC and any follow-up report shall be submitted to the U. S. Nuclear Regulatory Commission when authorization is sought to resume operation of the reactor.

The U.S. Nuclear Regulatory Commission (NRC) shall also be notified within the time frame required by the facility reporting requirements. The reactor shall not be restarted without NRC prior approval.

12.4.2 Reportable Events

Certain occurrences are defined to be reportable events. The extent of the action to be taken would depend on the event and the circumstances that exist at the time. However, as a minimum, written procedures direct that the reactor be shut down pending implementation of corrective action and senior management of the facility be notified. The occurrence shall be investigated, its root cause determined, and corrective action implemented, and the U.S. NRC shall be notified within the time frame required by the facility reporting requirements.

Occurrences that constitute reportable events include:

a) Any actual safety system setting less conservative than specified in the MITR Technical Specifications except during periods of instrument maintenance with the reactor shut down,

b) Operation in violation of a limiting condition for operation,

c) Safety system component malfunction or other component or system malfunction which could, or which threatens to, render the safety system incapable of performing its intended function,

d) Release of fission products from a fuel element in a quantity that would indicate a fuel element cladding failure,

e) An uncontrolled or unanticipated change in reactivity greater than 1.0% $\Delta K/K$,

f) An observed inadequacy in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor,

g) Conditions arising from natural or offsite manmade events that affect or threaten to affect the safe operation of the facility,

Additional reporting requirements are given in 10 CFR 20, Subpart M as well as 10 CFR 30.50(b) and 10 CFR 20.1906.

12.5 Reports

Three types of reports are submitted to the U.S. Nuclear Regulatory Commission. These are the facility's annual report, reportable occurrence reports, and special reports. The annual report provides information on facility operation. It is also used to satisfy the reporting requirements of 10 CFR 50.59 because it contains a summary of the changes made to the facility and procedures as well as tests and experiments conducted under the authority of 10 CFR 50.59. Reportable occurrence reports are the documents that result from the investigation of any of the occurrences listed in Section 12.4.2 of this report. Special reports, which are referred to as items of information, notify the NRC of changes of facility management or other similar administrative items. All reports are submitted to the NRC Document Control Desk. Copies will be provided in accordance with the current NRC instruction on internal organization.

12.5.1 Annual Report

The annual report will be submitted within ninety days following the 31st of December of each year. Its content is as follows:

a) A narrative summary of operating experience (including experiments performed) and of changes in facility design, performance characteristics and operating procedures related to reactor safety that occurred during the reporting period, as well as results of surveillance tests and inspections required by the MITR Technical Specifications.

b) Tabulation showing the energy generated by the reactor, the number of hours the reactor was critical, and the cumulative total energy output since initial criticality.

c) The number of emergency shutdowns and inadvertent scrams, including the reasons therefore.

d) Discussion of the major maintenance operations performed during the period, including the effect, if any, on the safe operation of the reactor, and the reasons for any corrective maintenance required.

e) A description of each change to the facility or procedures, tests, and experiments carried out under the conditions of 10 CFR 50.59 including a summary of the safety evaluation of each.

f) A description of any environmental surveys performed outside the facility.

g) A summary of radiation exposures received by facility personnel and experimenters, including the dates and time of significant exposures, and a summary of the results of radiation and contamination surveys performed within the facility.

h) A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the licensee as measured at or prior to the point of such release or discharge.

(i) Liquid Waste (summarized on a monthly basis)

- Total gross radioactivity released during the reporting period, excluding tritium.

- Total tritium radioactivity and average concentration released during the reporting period.

- Total radioactivity (beta/gamma) released for specific nuclides, if the gross beta radioactivity exceeds 1×10^{-5} µCi/ml at point of release, during the reporting period.

- Total volume of effluent water (including diluent) during periods of release.

(ii) Gaseous Waste

- Radioactivity of principal radionuclides discharged during the reporting period for
  - Gases
  - Particulates, with half lives greater than eight days.

- The effluent concentration limit used and the estimated activity discharged during the reporting period, by nuclide for principal radionuclides, based on representative isotopic analysis.

(iii) Solid Waste

- The total amount of solid waste packaged.

- The total activity and type of activity involved.

- The dates of shipment and disposition (if shipped offsite).

i) A summary of the use of the medical therapy facilities for human therapy.

(i) Investigative Studies

- Nature and status of the studies.
- Number of patients involved.

(ii) Human Therapy

- Number of patients treated.
- Type of cancer treated.

12.5.2 Reportable Occurrence Reports

Reportable occurrences are reported to the NRC within twenty-four hours by telephone. In addition, a written report is provided within ten working days. The content of the written report is as follows:

a) Time and date of occurrence.

b) Identification of occurrence.

c) Reactor status at time of occurrence.

d) Description of the occurrence.

e) Apparent cause of occurrence.

f) Analysis of the occurrence and safety implications, if any.

g) Corrective action.

h) Failure data.

12.5.3 Special Reports

Special reports, other than reportable occurrence reports, are provided as appropriate, and there is no prespecified content for these reports. However, such reports shall be submitted in writing within 30 days in the event of:

a) Permanent changes in the facility organization at the level of the Director of Reactor Operations or above, and

b) Significant changes to the transient or accident analysis as discussed in this Safety Analysis Report.

12.6 Records

12.6.1 Record Keeping System

Records of the activities listed below shall be retained for the specified intervals. Record keeping is the responsibility of the Quality Assurance Supervisor who reports directly to the Director of Reactor Operations. Records may be in the form of logs, data sheets, or other suitable forms including electronic data storage. If the latter is used, then a capability to read the storage medium shall also be maintained.

12.6.2 Records Retention

All records relative to the following areas shall be retained for five years:

a) Records of normal reactor operation including power levels and periods of operation at each power level.

b) Records of principal maintenance activities including inspection, repair, substitution or replacement of principal items of equipment pertaining to nuclear safety.

c) Records of reportable occurrences.

d) Records of surveillance activities that are required by the technical specifications.

e) Records of reactor facility radiation and contamination surveys.

f) Records of experiments performed with the reactor.

g) Records of fuel inventories, receipts, and shipments. (Note: Records of individual fuel element usage shall be retained until the element is returned to the U.S. Department of Energy.)

h) Records of changes made in the operating procedures.

i) Records of audit reports including both internal audits and those performed for or by the MITRSC.

All records relative to the following areas shall be retained for one certification cycle, which is currently six years:

j) Records of individual licensed staff members indicating qualifications, experience, training, and retraining. (<u>Note</u>: These are retained for six years or until the license is renewed or surrendered.)

All records relative to the following areas shall be retained for the life of the facility:

k) Records of radioactivity in liquid and gaseous wastes released to the environment.

l) Records of offsite environmental monitoring surveys.

m) Records of radiation exposure of all plant personnel and others who enter radiation control areas.

n) Records and drawing changes reflecting plant design modifications made to systems and equipment described in the Safety Analysis Report.

o) Records of radioactive shipments, including solid waste disposal.

12.7 Emergency Planning

The emergency plan for the MIT Research Reactor was written in accordance with NUREG-0849, Regulatory Guide 2.6 dated March 1983, and ANSI/ANS 15.16-1978. The plan was rewritten in its entirety in 1996 - 1997 and the rewritten version received final NRC approval in August 1998. The plan is reviewed annually.

The MITR emergency plan provides for a response to three of the four recognized categories of emergency. These are in ascending order of severity: notification of unusual events, alert, and site area emergency. The fourth category, general emergency, is not relevant to the MITR because the source term (6 MW reactor operating at atmospheric pressure and 55 °C maximum) is not capable of creating conditions that would warrant this class of emergency.

The sequence of actions that would be taken in the event of an emergency at the MITR are as follows:

a) The reactor is shut down.

b) The containment building is isolated.

c) Experimenters are evacuated.

d) Medical assistance, if required, is requested.

e) Emergency core cooling is connected but not initiated unless required.

f) The class of the emergency is determined, and civil authorities notified.

g) Off-duty licensed and radiation protection personnel are notified.

h) The MIT Campus Police are requested to stand by.

i) Radiation levels are monitored on-site and tracked off-site using the MITR Radiation Protection Office's remote monitors.

j) Radiation surveys by the MIT Campus Police are initiated if such surveys are required.

k) Off-duty personnel are briefed as they arrive.

The capability to seal the reactor containment building and the presence of passive safety features for decay heat removal (see Chapter 6 of this report) render it extremely unlikely that there would ever be a release of radioactive material during an accident.

Drills on various sections of the plan are held as part of both the initial qualification and requalification programs. An exercise that involves the participation of the MIT Campus Police and notification to local civil authorities is held annually.

12.8 <u>Security Planning</u>

A security plan has been prepared as required by 10 CFR 50.54(p)(2) and 10 CFR 73, and has been approved, as amended, by the U.S. Nuclear Regulatory Commission. This plan includes descriptions of Special Nuclear Material (SNM) use and storage areas, security devices, access control, security organizations, response procedures, and SNM transportation requirements. This plan is reviewed biennially.

12.9 Quality Assurance

The MITR quality assurance program encompasses both procedures and equipment important to safety. The program ensures that independent checks are performed on the creation of new procedures and the revision of existing procedures, as well as on the design, fabrication, and installation of equipment and/or systems important to safety. In addition, the program provides for complete and accurate information such as a means of tracing procedure changes and of generating "as-built" drawings of equipment.

12.9.1 Procedures

Both the creation of new procedures and the revision of existing ones are subject to a safety evaluation that is documented in a "safety review." This review includes:

a) A description of the proposed change including the rationale for it.

b) A safety analysis of the proposed change including verification that it will not adversely affect the emergency plan, security program, or requalification program.

c) Verification that an unreviewed safety question does not exist.

d) ALARA impact evaluation.

Safety reviews are required for all procedures important to safety. However, the level of the review depends on the nature of the procedure. Class B procedures require approval by two senior reactor operators and the Director of Reactor Operations. These include the administrative procedures, operating procedure checklists, abnormal operating procedures, technical specification tests, maintenance and special procedures involving nuclear safety, and preoperational tests of equipment important to safety. Class A procedures require a Class B review and approval by the MIT Committee on Reactor Safeguards. These include the standard operating plans, emergency plan, requalification program, and security plan. In addition, certain changes may require either (1) notification to the U.S Nuclear Regulatory Commission of the change with a determination that the change has no adverse impact, or (2) prior approval of that agency. Changes to the technical specifications and/or license require prior NRC approval.

All safety reviews are maintained for at least five years, and those that contain plant design modifications made to systems and equipment described in this report are retained for the life of the facility. The program is audited quarterly by the Quality Assurance Supervisor or designate and spot-checked annually by the independent auditor who is designated by the MIT Committee on Reactor Safeguards.

12.9.2 Equipment

Both the procurement of new equipment/instruments and/or the upgrade of existing items are subject to a quality assurance review. Such reviews normally include a safety analysis that is documented as a safety review. The Quality Assurance Supervisor in conjunction with the Superintendent of Operations and Maintenance and the Director of Reactor Operations specifies the quality assurance requirements for each program. Items that might be required include:

a) A safety analysis.

b) An update of the system description in training materials.

c) Certification of the materials or components to be used.

d) Welding certification if welds are required.

e) Design and as-built drawings.

f) Preoperational testing.

g) Maintenance procedures.

The requirements are summarized on a Q/A form that also indicates who, among the staff at the MITR, is qualified to verify that the requirement has been met. If the necessary expertise is not available in-house, then outside services are specified. At least two individuals independently check the performance of each requirement. Once the program is completed, it is reviewed by the Quality Assurance Supervisor, the Superintendent of Operations and Maintenance, and the Director of Reactor Operations.

Quality assurance records are maintained for at least five years. The program is audited annually by the Quality Assurance Supervisor and spot-checked by the independent auditor who is designated by the MIT Committee on Reactor Safeguards.

12.10 Operator Training and Requalification

The training of newly hired personnel is discussed in Section 12.1.5.1 of this report. Policies that are implemented to ensure that licensed individuals both retain and improve the knowledge and skills acquired during their initial training are described here.

12.10.1 Objectives of the Requalification Program

The requalification program is, at a minimum, intended to meet the requirements of 10 CFR 55.59(c). The program's objectives are to verify that licensed individuals remain proficient on routine operations, to refresh knowledge of operations that are performed infrequently, to provide for the timely review of facility and procedural changes by all licensed personnel, and to ensure improvement of any areas of performance weakness that are identified.

12.10.2 Organizational Structure

The requalification program is under the direction of the Requalification Program Coordinator. As shown in Figure 12-1, this individual reports directly to the Director of Reactor Operations. In addition, this individual works closely with the Superintendent of Operations and Maintenance, and the Training Coordinator. The latter is responsible for the initial training of newly hired personnel at both the reactor operator and the senior reactor operator level.

12.10.3 Requalification Schedule

The requalification program is conducted over an interval of two years. Upon completion, it is followed by successive two-year programs.

12.10.4 Lectures and Reviews

Preplanned lectures are given on a regular and continuing basis over the two-year interval of the program. Examples of the topics included in the series are:

a) Subcritical multiplication and reactor startup.

b) Reactor dynamics (step/ramp reactivity insertions).

c) Reactivity feedback effects.

d) General and specific plant operating characteristics.

e) Normal, abnormal, and emergency operating procedures.

f) Handling of radioactive samples.

g) Nuclear medicine procedures for neutron capture therapy.

h) Biological effects of radiation.

i) Radiation control and safety.

j) Engineered safety features.

k) Technical specifications

l) Instrumentation and control systems.

m) Nuclear and process safety system.

The MITR operates 24 hours per day, seven days per week. Personnel on the evening and night shifts therefore sometimes have difficulty attending lectures. Also, a person may be sick or on vacation when a particular topic is covered. This situation is mitigated as follows:

a) Most individuals rotate through the evening/night shifts. Hence, it is unlikely an individual will miss every lecture.

b) If possible, lectures are scheduled at shift turnover times so that both on-coming and off-going personnel can attend.

c) A given lecture can be offered twice.

In the event that an individual does miss a lecture, self-study will be substituted. The individual will be given the lecture materials and any relevant references. A deadline for completion of the self-study (usually two weeks) will be specified. A written quiz will then be administered to verify comprehension. (See Section 12.10.6.3 of this report for grading criteria.)

12.10.5 On-the-Job Training

The objective of the on-the-job training aspect of the requalification program is to ensure that each licensed operator or senior operator (1) is involved in facility manipulations, (2) understands the operation of apparatus and mechanisms associated with control manipulations and knows the operating procedures, (3) is cognizant of changes in both facility design and procedures as well as the technical specifications, and (4) reviews the contents of all abnormal and emergency procedures on a regular basis. Reactor operators should perform and senior operators should perform or supervise routine operations such as startups, shutdowns, and significant power changes on a regular basis.

The following are the minimum requirements for on-the-job training over the two-year interval of the MITR program:

a) Each licensed individual shall perform at least ten reactivity control manipulations in any combination of reactor startups, shutdowns, or significant reactivity changes including reshims. The maximum time interval between manipulations shall not exceed four months.

b) Each licensed operator shall perform at least one startup in each two-year cycle. Each licensed senior operator shall either perform or supervise at least one startup in each two-year cycle for the program.

c) All licensed personnel shall read and initial or receive a copy of all safety reviews for the facility. Safety reviews document all changes to systems, procedures, and technical specifications. (Note: Documentation of this activity is done by the Quality Assurance Supervisor.)

d) All licensed operators shall receive training on emergency response at least once during the two-year cycle for the program. This could be done by:

    (i) Participating in the annual emergency plan exercise.

    (ii) Participating in the emergency plan drills that are held more frequently. (Note: These are the same as an exercise except that no off-site support agencies are involved.)

(iii) Participating in a walk-through of the facility related to proper emergency response.

e) All licensed personnel shall review both the abnormal and the emergency operating procedures on an annual basis. The performance of this review shall be documented.

f) Each licensed individual shall be observed and evaluated at least once per year on his or her response to an actual or simulated emergency or abnormal condition in which the proper use of an abnormal operating procedure is demonstrated.

12.10.6 Examinations

A comprehensive written requalification examination shall be given at least biennially to all licensed personnel except those (maximum of three) who are involved with exam preparation, administration, or grading.

12.10.6.1 Content

The examination will include questions taken from the following general areas:

a) Principles of reactor operations.

b) Features of facility design.

c) General operating characteristics of the reactor.

d) Instrumentation and controls.

e) Safety and emergency systems.

f) Standard and emergency operating procedures.

g) Radiation control and safety.

h) Reactor theory.

i) Radioactive materials handling, disposal, and hazards.

j) Specific operating characteristics of the reactor.

k) Fuel handling and core parameters.

l) Administrative procedures, conditions, and limitations.

The questions formulated for this examination will be of a complexity at least equal to the questions given at the MITR during the initial licensing exams that are administered by the U.S. Nuclear Regulatory Commission. Questions may be in a multiple choice, essay, or calculational format.

12.10.6.2 Conduct

This examination shall be closed-book, and it shall be proctored. (Note: Standard reference material such as a Chart of the Nuclides or Emergency Action Level tables may be provided if appropriate.)

12.10.6.3 Grading

The following criteria apply to all examinations given as part of the requalification program:

a) The passing criterion is 80% overall and for exams with multiple sections, 80% in each section. If such grades are achieved, no further remedial training is required. However, the results of all missed questions shall be reviewed.

b) A score in the range of 65 to 79 percent overall (or on any section) requires additional training on those areas or topics where weaknesses or deficiencies are indicated. This retraining and retesting are to be completed within 60 days from the date the examination was graded and before the candidate is considered requalified. In this case, the candidate need not be removed from licensed duties subject to the evaluation of the Senior Review Board.

c) A score of less than 65 percent overall (or on any section) requires that an evaluation be performed by the Senior Review Board within 30 days of the date that the examination was graded. The evaluation is to determine if the deficiencies require that the individual be removed from licensed duties pending completion of any accelerated training. In any case, the licensed operator must pass a reexamination within four months of the administration of the initial exam or be removed from licensed duties.

d) Regardless of the score, if the individual's test indicates a deficiency in a critical area that affects safety, training is promptly administered to correct the deficiency or the operator is removed from licensed duties in the affected area until the deficiency is corrected.

Grading of the annual requalification exam is to be completed within thirty days of administration. Thus, the maximum time from exam administration to completion of retraining and retesting (including grading) for individuals in category (b) above is 90 days and for individuals in category (c) above is 60 days. (Note: The four month deadline for removal from licensed duties if upgrading is not successful starts from the date of the administration of the annual exam.)

12.10.7 Annual Walkthrough Examination

Each licensed operator and senior operator shall be given an annual walkthrough examination that is administered by either a member of the Senior Review Board or a senior reactor operator designated by that Board. The object of this walkthrough is for the examinee to demonstrate satisfactory understanding of the operation of facility systems and operating procedures. Particular emphasis should be placed on recent changes to the facility, its procedures, and/or the license.

The performance of these walkthroughs shall be documented.

12.10.8 Medical Certification

All licensed individuals shall have a medical examination at least once during each two-year interval of the requalification program. These examinations shall be conducted by a licensed physician. This physician would normally be a member of the MIT Medical Department and records of the examination would be held by that department. However, other arrangements may be made.

In the event that a licensed individual has or develops a medical condition that could interfere with the performance of licensed duties, notification shall be made by the facility to the U.S. Nuclear Regulatory Commission. No one whose physical condition is not in accordance with 10 CFR 55.57(b)(1) shall be allowed to perform licensed duties.

12.10.9 Requalification Records

Records shall be maintained to show compliance with the requirements of the training and requalification programs:

a) Initial Training: The required record is the completed, signed qualification card. This documents the satisfactory completion of all study modules and practical factors as well as the review of the candidate by the Senior Review Board.

b) Requalification: Each licensed operator and senior operator has an individual file that contains all documents related to initial training and requalification. Those records include:

   (i) Lectures attended and/or self-study.

   (ii) Reactivity manipulations (type, date, and time).

   (iii) Time spent acting as an operator or senior operator (date and hours).

   (iv) Startups performed (date and time).

   (v) Emergency plan training.

   (vi) Reviews of the abnormal/emergency operating procedures.

   (vii) Response to an actual or simulated abnormal condition.

   (viii) Certification of medical examination.

   (ix) Written examination results.

   (x) Annual walkthrough results.

   (xi) Upgrading, if any.

   Documentation that all licensed personnel have read all safety reviews is contained in the safety review files. Medical examination results are held by the MIT Medical Department. The above records are retained for at least six years or until the license of the individual in question is renewed or surrendered.

c) Master Requalification File: Master files are kept of the schedule of all lectures, lecture outlines, attendance sheets, exams, and other material necessary to implement the requalification program.

12.10.10 Maintenance of Active Status

In order to maintain active status, each licensed operator or senior operator shall actively perform the functions of reactor operator or senior reactor operator for a minimum of four hours each calendar quarter. For senior reactor operators, direct supervision of these operations may be considered equivalent to actual performance. If this requirement is not met, the license becomes inactive. In order to reactivate an inactive license, the individual shall perform at least six hours of licensed activities in the position that is being recertified under the direction of a licensed operator or senior reactor operator as appropriate.

12.10.11 Training Program Audit

The training program (initial training and requalification) is one of the items audited by the independent auditor who is designated by the MIT Committee on Reactor Safeguards. This is done annually.

Chapter 13

Accident Analysis

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Chapter 13

Accident Analysis

13.1 Accident-Initiating Events and Scenarios

A summary is provided here of accident-initiating events. Analysis of these events and determination of consequences is given in Section 13.2 of this report. One of the events listed below, the Maximum Hypothetical Accident, involves a scenario that causes fuel damage. None of the other events that are listed below result in core damage.

13.1.1 Maximum Hypothetical Accident

The maximum hypothetical accident (MHA) for the MITR is postulated to be a coolant flow blockage in the fuel element that contains the hottest fuel plate. It is assumed that the entire active portion of four fuel plates melts. This accident could occur if a foreign object were to fall into the core tank and be undetected. In order to cause damage, the object would have to fall through the lower grid plate. This could only occur during a refueling when an element position was open. Fuel element nozzles have multiple openings to allow coolant flow. For the foreign object to block flow to the fuel plates, it would have to pass through one opening, thus putting an effective size limit on the object. Hence, a maximum of only four plates would be affected. If this accident were to occur, the following indications, automatic protective actions, and corrective actions would occur:

a) Boiling might occur in the blocked channels. If so, the formation and collapse of vapor bubbles would create high frequency noise that should be observed on the linear flux channels. MITR operators are directed by procedure to lower reactor power if this is observed.

b) If power is not lowered, then melting of the fuel plates is postulated to occur. Fission product gases would be detected by the core purge monitor which provides an alarm in the control room. MITR operators are directed by procedure to shut the reactor down if elevated core purge radiation levels exist.

c) If reactor operation is continued, the fission product activity would be merged with the building exhaust ventilation. The plenum radiation monitors would become elevated and alarm. Building ventilation would be tripped and the ventilation dampers would close, thereby sealing the building and stopping any release.

d) If the main ventilation dampers failed to close, the auxiliary intake and exhaust ventilation dampers would close, thereby sealing the building and stopping any release.

To summarize, there would be at least three distinct indications to the console operator for this accident. These are noise on the linear flux channels, elevated core purge radiation levels, and elevated plenum effluent monitor levels. Also, an engineered safeguards feature, the plenum monitor building isolation interlock, would automatically stop the release even if the operator failed to act.

13.1.2 Insertion of Excess Reactivity

Damage to the core from this type of accident is not considered credible because of both administrative controls and reactor design features. Initiating scenarios are:

a) Step Reactivity Insertion: An initiation event could be the failure of an in-core sample assembly or ICSA. For example, an ICSA thimble could fail open and the force of the coolant could lift the sample out of the facility thereby inserting positive reactivity. The MITR is protected against this type of occurrence by (1) careful design of all such facilities and (2) limitations on the allowed reactivity worth of all ICSA samples. Thus, even if an ICSA failed, the consequences could not create a step reactivity insertion greater than the maximum allowed.

b) Ramp Reactivity Insertion: An initiating event could be the overspeed of a shim blade drive. The MITR is protected against this type of occurrence as follows:

   (i) During normal operation, the shim blade speed is locked to the frequency of the electric power grid. Operation at other than the design speed is not possible.

   (ii) During the performance of digital control experiments, shim blade speed can be varied up to a maximum. Protection against overspeed is provided by use of a special trip on reactor period and the careful design of all such experiments.

13.1.3 Loss of Primary Coolant

Damage to the core from this type of accident is not considered credible because the core is contained in two concentric tanks, both of which would have to fail to cause a complete loss of primary coolant. Also, the reactor is equipped with an emergency core cooling system that would provide cooling to the fuel elements if a complete loss of coolant were to occur. Initiating scenarios are:

a) Break of Primary Coolant Piping: This is unlikely given the design of the piping. Were it to occur, the core would be unaffected because the reactor would scram on low primary flow and the anti-siphon valves would open to prevent the siphoning of coolant from the core tank.

b) Break in Core Tank: The core would remain covered because the reactor is contained in two concentric tanks, the light-water core tank and the heavy-water reflector tank. Light water and heavy water would mix and cause a negative insertion of reactivity.

c) Sample Malfunction in Proximity to Core Tank: Administrative procedures are implemented to exclude samples that could cause any damage to the core tank during an irradiation.

13.1.4 Loss of Primary Coolant Flow

Damage to the core from this type of accident is not considered credible. A loss of primary coolant flow could (and has) resulted from a loss of normal electrical power and/or a failure of a primary coolant pump. Both events cause the reactor to shut down automatically and natural convection cooling is established and removes the decay heat.

13.1.5 Mishandling or Malfunction of Fuel

Mishandling of fuel is unlikely because of the careful design of the fuel handling tools and because of written schedules that are followed whenever fuel is to be handled. Incipient fuel element clad failures have occurred during normal reactor operation. They are quickly detected by the core purge monitoring system. The corrective action is to shut the reactor down, to identify the fuel element in question, and to remove it from the core. This action occurs well before any 10 CFR 20 limit is approached.

13.1.6 Experiment Malfunction

Damage to the core from this type of accident is not considered credible because of the careful design of all in-core experiment facilities and the administrative limits imposed on all experiments, in-core and ex-core.

13.1.7 External Events

Damage to the core from external events (lightning, floods, meteorological disturbances, and seismic events) is not considered credible because the core is contained within two concentric tanks and a full containment building.

13.1.8 Mishandling or Malfunction of Equipment

Damage to the core from accidents of this type is not considered credible because of the reactor's passive safety features. The reactor is designed so that instrument or equipment failures generally result in a reactor shutdown. Also, strict administrative and procedural controls are implemented for situations for which it is not possible to devise interlocks and/or a fail-safe response.

13.2 Accident Analysis and Determination of Consequences

The MITR is one of a number of research reactors that are operated across the United States by universities, government laboratories, and private industry. Research reactors constitute a distinct class of reactor. Other classes are test, power, military, and space craft. Each reactor class is subject to certain restrictions on its design and operation. For research reactors, these are that the power level be no more than 10 MW, that in-core experiments have a cross-sectional area of less than sixteen square inches, and that there be no fueled loops. These restrictions result in a type of reactor that is suitable for siting on university campuses where they can be used for both teaching (physics laboratories, reactor physics demonstrations, radiological health education, etc.) and research (nuclear medicine and cancer therapy, identification of air pollutants, archaeological studies, etc.).

In addition to the above noted general characteristics of research reactors, the MITR has several other features that greatly limit its potential for creating accidents, especially accidents with off-site consequences. These are:

a) The fuel is a cermet which is made by sintering so that the final product contains a certain void fraction. (See Section 4.2.1 of this report.) These voids retain fission product gases. Hence, should there be a clad failure, the fission product gas must diffuse through the sintered fuel matrix in order to escape.

b) The reactor core is contained within a closed primary system. This system is not airtight, but it would retard the release of fission products should there be a clad failure. (See Sections 4.3 and 5.2 of this report.)

c) The entire reactor is contained in a gas-tight containment building that is sealed automatically should effluent radiation levels be abnormal. (See Section 6.5 of this report.)

The absence of a significant source term, the lack of a driving force for fission product release, the use of a cermet fuel design, the presence of a closed primary system, and the use of a full containment minimize the potential for any accident at the MITR to have off-site radiological consequences.

13.2.1 Maximum Hypothetical Accident

The maximum hypothetical accident (MHA) for the MITR is postulated to be a coolant flow blockage in the fuel element that contains the hottest fuel plate. This could occur as the result of some foreign material falling into the core tank during a refueling. After the primary pumps are started, the object would be swept from the bottom of the tank up to the fuel element nozzle so that flow to the fuel plates was obstructed. In order for this to happen, the foreign material would have to fall through the lower grid plate. This could only occur during a refueling, when a fuel element was removed so that the corresponding position was open. The size of the openings in the lower grid plate would restrict the dimensions of the foreign object to those of a fuel element nozzle. Coolant can pass through either a nozzle's end or side openings. Hence, the foreign object could not block all coolant flow through the nozzle. However, if the material were small enough to enter the triangular entrance in the nozzle, it might possibly block the flow to a maximum of five coolant channels (six plates). Because the two fuel plates on the outer regions of the blocked area will be cooled from one side, the only melting that might occur would involve the inner four fuel plates. It is conservatively assumed that all five coolant channels are blocked and that the entire active portion of the four associated plates melts completely. This is a very conservative assumption because the coolant channels can only be partially blocked because of the geometry of the nozzle. Experience with fuel plate melting both at the Materials Testing Reactor (MTR) and at the Oak Ridge Research Reactor has shown that fuel plate melting because of flow blockage does not propagate beyond the affected flow channels. Although nearby plates were discolored, cooling by the unaffected channels was sufficient to prevent propagation of the melting [13-1, 13-2].

An analysis of fission product release and radiation dose to the off-site population was previously performed by Mull for the MITR-II [13-3]. That analysis was recently redone by Li both for higher reactor powers and with an updated source term [13-4]. In both analyses, the fission products in the fuel at the time of the accident were assumed to be in equilibrium for the steady-state reactor power. This assumption is conservative for the MITR because the reactor is shut down periodically for maintenance and refueling. Table 13-1 lists the equilibrium fission product activity inventories for reactor powers up to 10 MW. The fission product inventory for 6 MW is used for the current analysis.

13.2.1.1 Containment Source Term

Table 13-2 lists the fission product release fraction from the melted fuel (F_f), the fraction released from the primary coolant system (F_p), and the fraction remaining airborne in the containment atmosphere (F_c). These values, which are used in the current analysis, were taken from Li [13-4] except that the primary coolant system release fraction (Fp) was obtained from a coolant evaporation calculation. Mull [13-3] and Li [13-4] assumed primary coolant release fractions of 0.1 and 0.3 respectively. These values were chosen based on power reactor accident scenarios that involved severe accidents initiated from coolant system failure that eventually lead to core melt [13-5]. The MITR's MHA, which is initiated by coolant channel blockage, does not involve a primary coolant system failure. However, the primary coolant system is not leak tight. Hence volatiles, such as noble gases, may be released to the containment. Also, non-volatile fission product transport from the core tank to the containment is possible through coolant evaporation from the reactor core tank. (Note: Both the loss of volatiles and the evaporative release mechanism for non-volatiles ignore the presence of the reactor top shield lid which is required to be in place if reactor power exceeds 100 kW. The presence of this lid makes the primary system a barrier to fission product release.) A calculation was made to estimate the amount of fission product release to the containment through evaporation during a two hour period [13-6]. Assumptions for this calculation are that the coolant temperature is 60° C, the relative humidity in the upper core tank air space is 10%, the temperature in the air space is 20° C, and the fission products mix uniformly with the primary coolant in the core tank (about 700 gallons). The first assumption specifies the highest possible coolant temperature (LSSS), the second and third assumptions establish the lower bound of air conditions which would result in a higher evaporation rate, and the last assumption conservatively uses only the coolant volume in the core tank instead of the total coolant volume in the primary coolant system. This calculation shows that about 1.6 gallons of

Table 13-1

MITR Core Fission Product Inventory [13-4]

Isotope		Half-life	λ_i(sec-1)	Y_i(%)	Q_s^i ($\times 10^5$ Ci)					
					5MW	6MW	7MW	8MW	9MW	10MW
Kr	85m	4.36h	4.41E-5	1.5	0.6490	0.7788	0.9086	1.0384	1.1682	1.3000
	87	78m	1.48E-4	2.7	1.1700	1.4040	1.6380	1.8720	2.1060	2.3400
	88	2.77h	6.95E-5	3.7	1.6000	1.9200	2.2400	2.5600	2.8800	3.2000
Xe	131m	12.0d	6.68E-7	0.03	0.0130	0.0156	0.0182	0.0208	0.0234	0.0260
	133m	2.3d	3.49E-6	0.16	0.0692	0.0830	0.0969	0.1107	0.1246	0.1380
	133	5.27d	1.52E-6	6.5	2.8100	3.3720	3.9340	4.4960	5.0580	5.6200
	135m	15.6m	7.40E-4	1.8	0.7780	0.9336	1.0892	1.2448	1.4004	1.5600
	135	9.13h	2.11E-5	6.2	0.4130	0.4956	0.5782	0.6608	0.7434	0.8260
	138	17m	6.79E-4	5.5	2.3800	2.8560	3.3320	3.8080	4.2840	4.7600
I	131	8.05d	9.96E-7	2.9	1.2500	1.5000	1.7500	2.000	2.2500	2.5100
	132	2.4h	8.02E-5	4.4	1.9000	2.2800	2.6600	3.0400	3.4200	3.8100
	133	20.8h	9.25E-6	6.5	2.8100	3.3720	3.9340	4.4960	5.0580	5.6200
	134	52.5m	2.20E-5	7.6	3.2900	3.9480	4.6060	5.2640	5.9220	6.5700
	135	6.68h	2.89E-5	5.9	2.5500	3.0600	3.5700	4.0800	4.5900	5.1000
Br	83	2.4h	8.02E-5	0.48	0.2080	0.2496	0.2912	0.3328	0.3744	0.4150
	84	30m	3.85E-5	1.1	0.4760	0.5712	0.6664	0.7616	0.8568	0.9510
Cs	134	2.0y	1.10E-8	0.0*	2.8600	3.4320	4.0040	4.5760	5.1480	5.7200
	136	13d	6.17E-7	0.006*	0.4140	0.4968	0.5796	0.6624	0.7452	0.8280
	137	26.6y	8.27E-10	5.9	2.3100	2.7720	3.2340	3.6960	4.1580	4.6200
Rb	86	19.5d	4.11E-7	2.8E-5*	0.6120	0.7344	0.8568	0.9792	1.1016	1.2200
Te	127m	90d	8.82E-8	0.056	0.0242	0.0290	0.0339	0.0387	0.0436	0.0484
	127	9.3h	2.07E-5	0.25	0.1080	0.1296	0.1512	0.1728	0.1944	0.2160
	129m	33d	2.43E-7	0.34	0.1470	0.1764	0.2058	0.2352	0.2646	0.2940
	129	72m	1.60E-4	1.0	0.4320	0.5184	0.6048	0.6912	0.7776	0.8650
	131m	30h	6.42E-5	0.44	0.1900	0.2280	0.2660	0.3040	0.3420	0.3810
	131	24.8m	4.66E-4	2.9	1.2500	1.500	1.7500	2.000	2.2500	2.5100
	132	77h	2.50E-6	4.4	1.9000	2.2800	2.6600	3.0400	3.4200	3.8100
	133m	63m	1.83E-4	4.6	1.9900	2.3880	2.7860	3.1840	3.5820	3.9800
	134	44m	2.63E-4	6.7	2.9000	3.4800	4.0600	4.6400	5.2200	5.8000
Sr	91	97h	2.99e-5	5.9	2.5500	3.0600	3.5700	4.0800	4.5900	5.1000
Ba	140	12.8d	6.27E-7	6.3	2.7200	3.2640	3.8080	4.3520	4.8960	5.4500

Table 13-1 (cont'd)

MITR Core Fission Product Inventory [13-4]

Isotope		Half-life	λ_i(sec-1)	Y_i(%)	Q_s^i ($\times 10^5$ Ci)					
					5MW	6MW	7MW	8MW	9MW	10MW
Ru	103	41d	1.96E-7	2.9	1.2500	1.5000	1.7500	2.0000	2.2500	2.5100
	105	4.5h	4.28E-5	0.9	0.3890	0.4668	0.5446	0.6224	0.7002	0.7790
	106	1.0y	2.20E-8	0.38	0.1640	0.1968	0.2296	0.2624	0.2952	0.3290
Rh	103	36.5h	5.27E-6	0.9	0.3890	0.4668	0.5446	0.6224	0.7002	0.7790
Tc	99m	6.04h	3.19E-5	0.6	0.2590	0.3108	0.3626	0.4144	0.4662	0.5190
Mo	99	67h	2.88E-6	6.1	2.6400	3.1680	3.6960	4.2240	4.7520	5.2800
Sb	127	93h	2.07E-6	0.25	0.1080	0.1296	0.1512	0.1728	0.1944	0.2160
	129	4.6h	4.32E-5	1.0	4.3200	5.1840	6.0480	6.9120	7.7760	8.6500
Nd	147	11.3d	7.10E-7	2.6	1.1200	1.3440	1.5680	1.7920	2.0160	2.2500
La	140	40.2h	4.79E-6	6.3	2.7200	3.2640	3.8080	4.3520	4.8960	5.4500
Ce	141	32d	2.51E-7	6.0	2.5900	3.1080	3.6260	4.1440	4.6620	5.1900
	143	32h	6.01E-6	6.2	2.6800	3.2160	3.7520	4.2880	4.8240	5.3600
	144	290d	2.76E-8	6.1	2.6400	3.1680	3.6960	4.2240	4.7520	5.2800
Zr	95	63d	1.27E-7	6.4	2.7700	3.3240	3.8780	4.4320	4.9860	5.5400
	97	17h	1.13E-5	6.2	2.6800	3.2160	3.7520	4.2880	4.8240	5.3600
Nb	95	35d	2.29E-7	6.4	2.7700	3.3240	3.8780	4.4320	4.9860	5.5400

Table 13-2

Fission Product Release Fractions

Fission Product	Fraction Released from the Melted Fuel F_f	Fraction Released from the Primary Coolant System F_p *	Fraction Remaining Airborne in the Containment Atmosphere F_c
Noble Gases	1.0	1.0	1.0
I	0.9	0.03	0.3
Cs	0.9	0.03	0.3
Te	0.23	0.03	0.9
Sr	0.01	0.03	0.9
Ba	0.01	0.03	0.9
Ru	0.01	0.03	0.9
La	0.0001	0.03	0.9
Ce	0.0001	0.03	0.9
Others	0.0001	0.03	0.9

* Based on coolant evaporation calculation.

primary coolant would be lost through evaporation during a two-hour period. This is equivalent to about 0.3% of the primary coolant in the core tank. The actual fraction would be lower because both the pool of water above the core and the presence of the reactor top shield lid would limit the release rate. A coolant system release fraction of 3% is adopted to conservatively bound the non-volatile fission product release. In addition, it is assumed that 100% of the noble gases are released to the containment.

Because the MITR has no containment spray or other engineered safeguards features to reduce the quantity of fission products in the containment atmosphere, depletion of the radioactive isotopes released to the containment occurs through natural processes. These include agglomeration and sedimentation. Agglomeration is the process by which the size distribution of airborne particulate tends to shift with time to larger sizes until an equilibrium condition is reached. This process affects sedimentation which is deposition because of gravitation. The noble gases are not expected to undergo either of these depletion process and thus they remain in the containment atmosphere.

The fission product activities in the containment atmosphere will vary with time. Activity initially increases as more fission products are released from the melted plates. A maximum occurs when a balance is reached with the depletion processes described above. The activity then starts to decrease because the natural depletion processes and leakage continue while the source is finite. It was assumed for the analysis that the containment activity was at its maximum (instantaneous release) from the beginning of the accident and the natural depletion processes started to take place simultaneously. In this analysis, fission product leakage from the containment was neglected as a removal mechanism. During the two hour period, it was assumed that the depletion for iodine and cesium was 70%, and that depletion for the other non-volatile elements was 10% [13-4].

13.2.1.2 Off-Site Radiation Dose Calculations

The following approaches were used to evaluate effects of the major release paths to the exclusion area during the maximum hypothetical accident:

a) An analysis was made to determine the atmospheric release from the containment building. The radiation doses that resulted from leakage (including external gamma dose from plume, beta dose, and thyroid dose) were calculated using a standard Gaussian diffusion model and local meteorological data.

b) Gamma radiation reaching the boundary area by direct penetration of the containment shell was calculated using standard shielding calculations. A Compton scattering model was developed and applied to photon scattering from the steel containment roof.

c) An analysis for radiation streaming was performed for the truck airlock which is the largest containment penetration.

13.2.1.3 Atmospheric Release

There are two paths for the fission products in the containment building to be released to the outside. One is a controlled release through the stack via the containment's pressure relief system. The other is containment leakage which is not controllable.

The containment building was designed to withstand internal pressures up to 2.0 psi above atmospheric and 0.1 psi lower than atmospheric. The building is normally maintained at a pressure slightly less than atmospheric in order to prevent out-leakage. If high radiation levels were detected by the plenum gas or particulate monitors, the building ventilation system's intake and exhaust fans would stop and both isolation dampers would close automatically. The maximum permissible leakage rate is 1% of the building volume per day per psi of overpressure. An integral air leakage test of the containment building is performed periodically to ensure that this criterion is satisfied. It is assumed conservatively in the containment leakage calculation that the containment pressure remains constant at 2.0 psig during the accident and that this results in a continuous release of the fission products to the environment at the maximum permissible leakage rate.

The containment building is equipped with a pressure relief system which consists of a blower, roughing filters, two high-efficiency absolute particulate air filters, and an activated charcoal filter for removal of elemental iodine (See Section 6.5.4.2 of this report.). The volumetric flow rate through this system was obtained from experimental data [13-7]. The fractions of radionuclides penetrating through the filters of the pressure relief system are: 100% of noble gases and bromine, 5% of iodine, 50% of all the others [13-4].

Atmospheric dispersion of a pollutant is primarily dependent on (1) meteorological conditions such as ambient temperature, wind speed, time of day, insulation, and cloud cover (atmospheric stability), and (2) pollutant stack emission parameters such as effluent velocity and temperature. The stability of the atmosphere is determined by the atmospheric thermal gradient, which is called the lapse rate. Neutral stability exists for a vertical temperature gradient of -1° C/100 meters. Unstable conditions with lapse rates greater than -1 °C/100 m add to the buoyancy of an emission, and stable conditions (lapse rates less than -1 °C/100 m) tend to inhibit downward vertical motion of the pollutant gases (plume). Dispersion from an elevated source (stack) is affected by the mixing and dilution of polluted gases with the atmosphere.

For a stack release, the maximum ground-level concentration in a sector may occur beyond the exclusion area boundary distance. Therefore, for stack releases, the atmospheric relative concentration values are calculated at various distances. Values of dispersion coefficients, which depend on the downwind distance and the atmospheric stability category, can be determined from the Pasquill curves [13-8] (a set of diffusion coefficient curves versus plume travel distance). In most references, the dispersion coefficients are given as a set of curves over the range of 10^2 to 10^5 meters. It is impossible to extrapolate these curves accurately to the range of the MITR's exclusion area distance, 8 to 21 meters. One alternative is to use the interpolation formulas for σ_y and σ_z developed by Briggs which fit the Pasquill curves [13-9].

The meteorological data needed for the atmospheric relative concentration calculation include wind speed, wind direction, and a measure of atmospheric stability.

The meteorological data used in this report were recorded at the Boston Station, MA 240BS 93-95. The wind speed data are expressed in the units of knots, where one knot equals 1853 meters/hour. The annual average wind speed for each stability category in the Boston area is listed in Table 13-3. It is shown that class D (neutral stability) is the most frequent stability condition, accounting for about 74% of the total events.

For release from the stack, the more unstable an atmospheric condition, the more a pollutant will be deposited in a shorter range with a higher concentration. In contrast, a more stable atmosphere would disperse the pollutant over a wider range and thus result in a lower concentration. From the meteorological data for the Boston area, it was found that the dose rates at 8 and 21 m are negligible based on class C, D, and E which account for most of the atmospheric conditions (frequency of 94%).

For containment release, the model ("exact" model) proposed in the U.S. NRC Regulatory Guide 1.145 is adopted [13-10]. Figure 13.1 shows the comparison of atmospheric relative concentrations for class A, B, C, D, E, and F versus distance. Class F represents a conservative estimate for both the site boundary and the restricted area and is therefore adopted as the limiting case of the ground release. It is noted that the calculated doses for class F stability would give a conservative estimate of the release with frequency greater than 99%. [13-4]

Table 13-3

Wind Speed for Each Stability Category (knots) Averaged Over All Directions

	A	B	C	D	E	F
N	0.0	5.4	7.7	10.3	7.2	4.8
NNE	0.0	6.1	8.2	11.0	6.3	4.5
NE	0.0	5.0	8.4	12.4	6.0	3.8
ENE	5.0	6.3	9.6	11.8	6.5	3.8
E	5.0	6.6	9.8	10.4	6.8	3.8
ESE	5.0	6.2	9.6	10.8	6.9	3.8
SE	4.5	7.1	8.4	9.4	6.3	4.1
SSE	5.0	5.8	7.3	9.0	6.3	4.4
S	1.0	5.0	8.5	10.6	6.6	4.8
SSW	4.5	5.6	9.1	12.1	7.4	5.1
SW	5.0	6.6	9.9	12.0	7.9	5.1
WSW	0.0	6.5	9.7	12.0	8.1	5.3
W	5.0	6.7	9.7	13.2	8.4	5.0
WNW	3.0	6.7	9.0	13.4	8.4	5.0
NW	5.0	6.1	10.0	13.2	8.3	5.0
NNW	4.0	6.5	9.0	12.5	8.2	4.6
Average	3.8	6.4	9.2	11.9	7.7	4.6
relative freq. (%)	0.00823	1.8254	8.3007	73.9423	12.0338	3.8154

* A-Extremely unstable, B-Moderately unstable, C-Slightly unstable, D-Neutral E-Slightly stable, F-Moderately stable.

13.2.1.4 Direct and Scattered Gamma Dose from Contained Source

Those radionuclides that are retained in the containment building constitute a source of gamma radiation. The gamma dose at the exclusion boundary consists of direct gamma dose, scattered gamma dose, and the gamma dose through the truck lock. Separate calculations were performed and the results summed for the two parts of the containment building (the sides which are shielded by both concrete and steel and the dome which is shielded only by steel as described in Section 6.5.1 of this report) and the truck lock. The truck lock is an eight-meter long rectangular steel passage closed at both ends by pneumatically sealed doors. Each door consists of a steel framework that is covered by steel plates on both sides. The two sides of the truck lock are shielded by concrete walls 0.5 meters thick while the front and top are not shielded. The radiation reaching the truck lock was treated as a point source located at the center of the inner surface of the inner door.

13.2.1.5 Conclusion for the Maximum Hypothetical Accident

A summary of the calculated results for the MITR MHA is given in Table 13-4. Even with the conservative assumptions of the release fractions and fission product equilibrium, the estimated maximum external doses to an individual located at the nearest point of public occupancy during the first two hours of the MITR MHA are 197 mrem at 8 m (back fence) and 247 mrem at 21 m (front fence) to the whole body. The maximum whole body dose is 300 mrem at 16 m. The internal doses are 135 mrem and 134 mrem to the thyroid for 8 m and 21 m, respectively.

Concentration of Ar-41 is predicted to be 1.79 x 10^{-3} µCi/ml for 6 MW. This estimate is extrapolated from measurements performed for the MITR at 5 MW. Compared to fission products released from the fuel, this concentration is lower by a factor of 5 to 7. Therefore, the contribution of Ar-41 to the off-site dose is negligible [13-4].

Table 13-4

Estimated Doses from all Modes of Radiation Release During a MITR Maximum Hypothetical Accident [13-4]

Component of the Dose	Dose (mrem) (c)	
	8 m (a)	21 m (b)
Whole body:		
Containment Leakage	12	12
Steel Dome Penetration	3	25
Shadow Shield Penetration	44	21
Air Scattering	57	75
Steel Scattering	87	114
Total (d)	197	247
Thyroid:		
Containment Leakage	135	134

(a) Boundary of restricted area

(b) Nearest point of public occupancy

(c) Calculation assumes that radiation emergency plan for protection of the public will be implemented in less than two hours.

(d) The maximum whole body dose is 300 mrem at 16 m.

13.2.2 Insertion of Excess Reactivity

In order for an insertion of excess reactivity to cause a reactor transient, it is necessary for the reactor to be at or near critical. The MITR core is designed with a mechanical interlock (the upper grid plate) that forces any operation that involves a large reactivity insertion to be done when the reactor is shut down. Specifically, access to the core is possible only if the upper grid is rotated, and that action is in turn possible only if all shim blades are fully inserted. Thus, refuelings can be performed only after the shim blades have been fully inserted. Also, in-core experimental facilities are fixed in place so that they too can be moved only when the grid is rotated. Excess reactivity transients are therefore unlikely because of the interlock between the shim blades and the grid itself. Hence, initiating events for such a transient are limited to control device malfunctions such as continuous withdrawals and/or the unanticipated movement of a sample contained within an in-core sample assembly.

Analyses of the MITR's response to step and ramp reactivity insertions indicated that the reactor could be shut down safely without damage to the core in the event of a transient up to a limiting value of either inserted reactivity (step) or a reactivity insertion rate (ramp). These values are the basis of both the step reactivity limit for any single experiment or component, which could credibly cause a reactivity effect by failure or ejection, and by the ramp reactivity insertion limit.

13.2.2.1 Step Reactivity Insertion

Step reactivity insertion analyses were performed for the MITR-II in the 1970s using a correlation derived from SPERT experiments [13-11]. The SPERT experiments span a range of core configurations sufficiently similar to the MITR-II core, such that the correlation was expected to give a good prediction of the MITR-II transient response. The step reactivity limit for the MITR-II core was determined using the derived correlation:

$$(\rho_L)_{Limit} = \frac{214.1c^{0.337}\ell^{0.663} + \beta_{eff}}{1 + 214.1c^{0.337}\ell^{0.663}} \tag{13-1}$$

where

c	is the magnitude of the negative void coefficient ($\Delta K/K/cm^3$),
ℓ	is the prompt neutron life time (seconds), and
β_{eff}	is the effective delayed neutron fraction.

The estimated step reactivity limit calculated using the above equation was 1.8% $\Delta K/K$ or 2.3 beta.

The MITR-III core is of the same geometry as that of the MITR-II. The step reactivity insertion limits for the MITR-III core were evaluated using the PARET code for both natural convection and forced convection cooling.

The step reactivity insertion limits for the MITR-III core were evaluated using the PARET code. The PARET code provides a coupled thermal, hydrodynamic, and point kinetics capability for research reactors under loss of flow and reactivity insertion transients [13-12]. Reactivity insertion transients were analyzed for research reactors utilizing both LEU and HEU cores using the PARET code by Woodruff [13-12]. Experimental data from the SPERT series of reactivity insertion tests have been used to benchmark the PARET code. For the HEU benchmark core, a step insertion of approximately 2.35 beta was found to be the limiting case, i.e., for reactivity insertions larger than this limit, the peak temperature of the cladding is predicted to exceed the clad melting temperature for Al-6061 alloy.

The initial power was assumed to be 5 W for natural convection and 6 MW for forced convection. It was assumed in the calculations that the clad surface is flat, because the PARET code does not provide the capability to model a finned surface. Therefore, it was conservatively assumed that the fins did not exist. The step reactivity is inserted within 1×10^{-3} second in these calculations. This is based on the reported intervals of 10^{-3} to 10^{-2} second for the SPERT experiments [13-13].

The most limiting case for a step reactivity insertion transient was identified to be one initiated at low reactor power and low coolant flow. These conditions result in a delay in the negative reactivity feedback that would mitigate any transients. Hence, under such initial conditions, there is more time for the reactor power to rise. This higher power and the lower heat transfer rate that results from low coolant flow lead to a higher fuel temperature. It was calculated that the maximum fuel clad temperatures are 531 °C for natural convection with a 2.0 beta step reactivity insertion and 360 °C for forced convection with a 3.0 beta step reactivity insertion. If the interval of the reactivity insertion is increased to 1.0 second, the limiting reactivity insertion is increased to 6 beta for natural convection [13-14].

The step reactivity insertion limits for the MITR-III are chosen to be 1.5 beta if natural convection exists and 2.3 beta if the forced convection flow is established before the reactor is taken critical. These figures correspond to 1.2% and 1.8% $\Delta K/K$. These figures are conservative given the above analyses.

13.2.2.2 Ramp Reactivity Insertion

For step reactivity insertions, the MITR's safety system had no role because the transient occurs rapidly enough that inherent feedback (void coefficient) terminates the excursion before the scram takes effect. Also, step insertions are more limiting if initiated from low power and low temperature because feedback effects are least under such conditions. Ramp reactivity insertions are quite different. The time scale is sufficiently long so that the MITR's safety system is the limiting factor. Accordingly, feedback effects were ignored in the analysis that follows. The MITR has a scram on both high power and short period. If a ramp insertion is initiated from low power, the period scram will shut the reactor down before the LSSS for power is approached. Hence, initiation of the ramp insertion from high power levels is the limiting case.

The analyses were performed by solving the space-independent kinetics equations for initial power levels of 6.0 MW and 80 kW [13-15]. The 6.0 MW value is the steady-state operating power level for the MITR-III and the 80 kW value is the maximum power level for natural convection operation (LSSS is 100 kW). For an initial power level of 6.0 MW, a ramp reactivity insertion rate of 83 mbeta/s (6.5×10^{-4} $\Delta K/K/s$) will cause power to reach the LSSS of 7.4 MW after 1.72 seconds. The reactor period will be slightly longer than 7.0 seconds. The calculation further assumed reactor shutdown occurred after a two second delay time even though the scram time for the MITR is less than one second. The reactor power reached 10.5 MW at the time of shutdown, which is lower than the safety limit of 11.2 MW at 2000 gpm.

From the above analysis, it is evident that if the reactor period scram is set at 7 seconds, it would provide a redundant protection for the scram on high power. The ramp insertion analysis was repeated using the same reactivity insertion rate for an initial power level of 80 kW. A period of 7.0 seconds was attained after 1.77 seconds. The power level at that time was 97.6 kW and the power level after a two second delay time was 140 kW. This is significantly lower than the corresponding safety limit of 350 kW [13-15].

The limiting reactivity insertion rate for the MITR-III is chosen to be 5×10^{-4} $\Delta K/K/s$. This figure is conservative given the above analysis, and provides adequate margin for purposes of operating the MITR.

13.2.2.3 Limitations on Excess Reactivity

Reactivity information on the MITR's shim blades and regulating rod is given in sections 4.2.2.5 and 4.5.1.5 of this report. As was noted in those sections, a refueled core normally attains criticality at a shim bank height of 7 to 9 inches. A typical value of the excess reactivity is about 6 beta. This is utilized to offset the effects of temperature, xenon, and fuel burnup. It should be noted that there are two design features that limit the potential possible impact of excess reactivity on the MITR. These are :

a) Subcritical Interlock
Observance of this interlock means that the infinite period shim bank height is at least 5 inches. This places an upper limit on the excess reactivity.

b) Shim Blade Withdrawal Limitation
There are six shim blades. It is not physically possible to withdraw more than one at a time. Hence, the excess reactivity per blade is quite modest.

13.2.3 Loss of Primary Coolant

A loss of primary coolant accident is not credible for the MITR for the reasons discussed in Chapters 4, 5, and 6 of this report. To summarize:

a) All penetrations to the light-water core tank occur well above the core. Hence, a break in the primary outlet piping could not uncover the core. (See Section 4.3 of this report.)

b) A break in the primary inlet piping could establish a siphon that would draw coolant through the core and up through the inlet plenum that is formed by the wall of the core tank and the core shroud. Such a siphon would be broken by the two redundant anti-siphon valves that are installed at the inlet penetration. (See Section 6.3 of this report.) Hence, a break in the primary inlet piping could not uncover the core.

c) A break in the core tank itself would not uncover the core because the light-water tank is wholly contained by the heavy-water reflector tank. (See Section 4.3 of this report.)

For the above reasons, a loss of coolant accident is not credible. Nevertheless, the MITR is equipped with an emergency core cooling system (ECCS) as described in

Sections 5.2.5 of this report. An analysis of the ECCS is given in Section 6.4 of this report. The ECCS is capable of removing the decay heat that would be present following extended operation at full power.

Analyses are given below of three events that were considered during the design of the MITR as possible initiators of a loss of coolant accident. In each case, the core remains both covered and properly cooled. Therefore, a complete loss of coolant accident is not considered to be credible.

13.2.3.1 Break in Primary Coolant Piping

A low-level core tank scram would occur when the coolant level drops four inches below the overflow level. If the break is in the outlet piping, the coolant level will drop until the outlet penetrations, which are about seven feet above the top of the core, are uncovered. If the break is in an inlet line, the coolant level will drop until the anti-siphon valves are uncovered. These are about six feet above the top of the core. In both cases, decay heat will be removed by natural circulation as described in Section 6.2 of this report.

13.2.3.2 Break in Light-Water Core Tank

Primary coolant would leak into the heavy water reflector where it would cause a strong negative reactivity effect as described in Section 4.5.1.6(c). The light-water coolant level would decrease to the height determined by the helium vent line from the D_2O reflector tank. This vent line returns to the basement at the same level as the primary coolant inlet and outlet pipes. (See Section 5.3.1.5 of this report.) The level of the coolant above the core would be at least 6 feet and decay heat would be removed by natural circulation. If the D_2O reflector dump system should be activated, then the light water would flow through the dump system and the vent lines to fill both the dump tank and the D_2O storage tank. The level of the coolant in the core tank would then drop to three or four feet above the core, depending on the volume in storage. Natural convection would still cool the core but the total heat capacity of the coolant in the core tank and its radiation shielding capability would be reduced and action to increase the water volume might be required.

13.2.3.3 Sample Malfunction in Proximity to Core Tank

The MITR contains a pneumatic tube in position 6RH1. This tube terminates inside a reflector tank reentrant thimble, thereby placing the sample in close proximity to the core tank. The consequences of a failure of this facility were examined by performing a planned failure within a laboratory mock-up. This mock-up consisted of the pneumatic system in a beam port adjacent to the core tank wall, all immersed in water to simulate the D_2O reflector. The MIT pneumatic tube rabbit is a polyethylene sample container with threaded cap ends. Samples may be sealed within an aluminum inner container. Two types of explosive failures were postulated. The first was the overpressurization of the polyethylene container and the second was the overpressurization of the aluminum container within the polyethylene container. Both accidents were simulated in an operating model. The first test, which used the polyethylene rabbit only, resulted in the end cap blowing off suddenly at 700 psi pressure. There was no evidence of damage to the pneumatic tube system. In the second test, the pressure reached 800 psi before the ends blew out of both the aluminum can and the rabbit. (This pressure rise is equivalent to about 280 mg of TNT.) Again, there was no evidence of damage to the pneumatic tube. Thus, in both experiments, damage was limited to the sample container and no measurable effect was found on the beam tube or the core tank wall. The potential energy that was released during these tests was greater than that which could possibly be generated by an approved sample. The hazard to the core tank is further reduced because any projectile must in turn go through the inner pneumatic tube, the outer pneumatic tube, the heavy-water reflector tank thimble and then reach the core tank wall.

13.2.4 Loss of Primary Coolant Flow

The MITR is designed so that a low primary flow (less than 1800 gpm or higher) will automatically initiate a scram. There are two initiating events that can cause a loss of primary coolant flow accident. The first is a loss of off-site electrical power which will stop the primary pumps and scram the reactor by dropping all six shim blades simultaneously. This is a credible scenario. The second is a pump coast down accident that occurs because of primary pump power supply failures or malfunctions of the pump motors. This is not considered to be a credible accident because the probability for both pumps to fail at the same time is very small.

The reactor will shut down automatically upon receiving a low primary coolant flow scram signal. The instrument delay time is less than one second.

An analysis was performed for a loss-of-flow accident subject to the assumption that the power supply to both primary coolant pumps was interrupted at the same time. The initial conditions and assumptions made in this analysis were as follows:

a) The steady-state equilibrium reactor power was 6.1 MW. This is the maximum steady-state power compatible with an LSSS of 7.4 MW and a primary coolant flow of 2000 gpm. It was chosen because the loss of flow is more severe if initiated from high power.

b) The primary coolant outlet temperature was 55 °C (scram setpoint).

c) The hot channel received the minimum flow among all coolant channels. This conservative assumption simplified the calculation by eliminating the need for analyzing individual coolant channels.

d) The axial power distribution was assumed uniform in both the average and the hot channel. (See Section 4.6.7 of this report.)

e) The steady-state primary coolant flow rate was 2000 gpm. The coast-down curve is shown in Figure 13-2.

f) The reactor scram was activated one second after the primary flow reached 1800 gpm. It is assumed that it would take one second for the shim blades to drop to 80% fully in position. The reactor power decreased linearly from 100% to 6.6% during the one second delay time.

The calculation was performed using the MULCH-II code, which was developed at MIT for the thermal hydraulic analysis of the MITR [13-16]. The code was benchmarked using experimental data on pump coast-downs obtained during the MITR-II initial startup testing [13-17]. The primary flow rate through the core region during the transient is shown in Figure 13-2. Figure 13-3 is a comparison of the coolant outlet temperatures for the average and the hot channels with different hot channel factors. A hot channel factor of 2.0, which was used in the thermal-hydraulic limit calculations given in Section 4.6 of this report, sets the upper bound for all possible operating conditions. The coolant temperature holds at the saturation value of 107.5 °C after 28.5 seconds into the transient because bubbles start to form. Figure 13-4 is a comparison of the fuel temperatures at the top of the fuel for the average and the hot channel with different hot channel factors.

13.2.5 Mishandling or Malfunction of Fuel

13.2.5.1 Mishandling of Fuel

The tools used for the handling of MITR fuel are described in Section 9.2.3 of this report. These tools are designed to prevent damage, such as scratches, to the clad during fuel movement. Also, they are designed to prevent the inadvertent drop of an element. These tool designs have been used since 1974 and no fuel damage has occurred during handling. If damage were to occur, or if it were suspected to have occurred, the element in question would not be utilized until an evaluation was performed. This could probably be done by visual inspection (through water for shielding) and, if felt necessary, by element sipping. The latter process entails placing the element in an approved storage location, such as the fuel storage ring, where it is immersed in water that is not subject to forced convection. If damage has occurred, fission products such as iodine will accumulate in the water. A sample of this water can be drawn after several days and counted for activity. Water from several control elements (ones with no suspected damage) would also be drawn for comparison.

Mishandling of MITR fuel would not cause a sudden radiological problem because the fuel matrix is a cermet, which acts to contain fission products. Fission products must diffuse through the fuel matrix in order to be released.

13.2.5.2 Malfunction of Fuel

Operating experience with MITR fuel is summarized in Section 4.2.1 of this report. A few plates, almost all from a former supplier, were identified as showing incipient excess outgassing. This was determined from the radioactivity in the purge gas (air) that is drawn across the core outlet plenum as described in Sections 5.2.1.7 and 9.1.5.2 of this report. Confirmation of the problem and identification of the fuel element involved was done through use of the fuel sipping method described above. Once confirmation was made, the element that contained the affected plate was removed from service. Malfunction of fuel does not engender serious radiological consequences because the MITR is equipped to identify any such malfunction in its incipient stages. Also, MITR operation does not impose significant thermal stress on the fuel. There is no mechanism (such as water-logging followed by a pulse) whereby an incipient condition could suddenly worsen. Specifically, the MITR does not operate in a pulse mode and the temperature change of the fuel from cold shutdown to hot operating conditions is about 30°C.

13.2.6 Experiment Malfunction

Experimental facilities and the review process for the use of these facilities is discussed in Chapter 10 of this report. From the perspective of reactor safety, these facilities may be divided into three groups. These are:

a) Ex-Reflector Facilities: These facilities are decoupled from the reactor so that they cannot affect either the thermal-hydraulic or the neutronic performance of the core. Examples include the medical irradiation rooms, any equipment monitored at the exterior of a beam port, pneumatic tubes that terminate in the graphite reflector, the sample processing areas, the in-reflector (graphite) irradiation facilities, the reactor floor hot cell, and the gamma irradiation facility.

b) In-Reflector Facilities: These facilities are in sufficient proximity to the reactor so that administrative limits are warranted to ensure that a malfunction could not affect the core. The principal facility in this category is the 2PH1 pneumatic tube that terminates in the 6RH1 reentrant thimble. An analysis of a malfunction of a sample in this tube is given in Section 13.2.3.3 of this report. That analysis showed that a severe overpressurization would not cause damage to the reactor. Nevertheless, administrative limits have been imposed on samples allowed in this (and all other) facilities. These address behaviors such as explosive energy, corrosion potential, radiolytic decomposition, and internal heating.

c) In-Core Facilities: These facilities are within the core itself. A fundamental principal of their design is that, despite their location, they be decoupled to the maximum extent possible from the core. Thus, in-core sample assemblies are often designed with a thimble that eliminates or restricts thermal interaction. Also, ICSAs often have dedicated cooling. Some coupling, such as neutronic, is, of course, unavoidable. ICSAs have been designed and installed in the MITR to do the following:

    i) Serve as a sample irradiation facility.

    ii) Serve as a temperature-controlled irradiation facility.

    iii) Replicate Pressurized Water Reactor (PWR) conditions to allow study of coolant chemistries for dose reduction.

    iv) Replicate Boiling Water Reactor (BWR) conditions to allow study of coolant chemistries for dose reduction.

    v) Replicate conditions that would result in Intergranular Accelerated Stress Corrosion Cracking (IASCC) and allow in-pile mechanical property testing.

Each of these facilities has been the subject of its own Safety Evaluation Report (SER) which documents the design, operation, safety, and accident potential of the facility. Appendix A to this Chapter is a condensed version of the SER for one of these ICSAs, the Boiling Coolant Chemistry Loop (BCCL). The BCCL SER is illustrative of the issues addressed for all in-core facilities.

13.2.7 Loss of Normal Electrical Power

The MITR was designed with the expectation that there would be interruptions of offsite electrical power. Passive safety features exist so that on loss of electricity the reactor will shut down. Specifically,

a) The six shim blades are attached to their drives by electromagnets. On loss of power, the magnets deenergize and the blades drop into the core under influence of gravity.

b) The air pressure that keeps the heavy-water reflector's dump valve closed is in turn controlled by an electrically-operated solenoid valve. On loss of electrical power, it opens thereby dumping the reflector and shutting the reactor down.

The MITR is also designed for the removal of decay heat by natural convection cooling. This is Mode #1 of the four modes of emergency cooling that are described in Section 5.2.5 of this report. Finally, both the main and auxiliary containment building ventilation dampers close on loss of offsite power, thereby precluding the release of any radioactive effluent to the environment.

The MITR is also equipped with an emergency power system as described in Section 8.2 of this report. Its principal purpose is to provide power for reactor and radiological instrumentation, auxiliary pump MM-2 which can be used for decay heat removal and lighting. None of these functions is essential to safety because:

a) The MITR's power level, coolant temperature (core outlet plenum), and core tank level are available either from battery-operated devices or instruments that do not use electricity.

b) Information on radioactive effluents is not required because the building is sealed (dampers closed).

c) Decay heat removal is achievable via natural convection within the core tank.

d) Battery-operated lights provide sufficient illumination for safety of personnel.

13.2.8 External Events

The MITR is protected from the impact of external events by the containment building, the design of which is given in Section 6.5 of this report. To summarize, it is a domed cylindrical structure with a diameter of 22.5 m (74 feet) and a height between grade elevation and the top of the dome of 14.9 m (49 feet). The shell is constructed of steel plate that is 9.5 mm (3/8 inch) thick on the sides and 15.8 mm (5/8 inch) thick on the dome. A cylindrical concrete wall, 2.0 feet thick and 31.5 feet high, is contained within the shell. The building is gas-tight and can withstand an internal pressure 2 psi above atmospheric pressure and an external pressure differential corresponding to -0.1 psi.

13.2.8.1 Lightning

The steel containment shell is grounded to a heavy copper conductor that is buried below the natural water table. Lightning arresters are attached to the ventilation exhaust stack and are also grounded to the buried copper conductor. Hence, lightning does not pose a direct threat to the facility. It could cause a loss of offsite electrical power. However, for the reasons discussed in Section 13.2.7 of this report, such a loss will not initiate a safety challenge.

13.2.8.2 Floods

The MITR is located in Cambridge, Massachusetts which lies in a basin formed by a low-lying chain of hills. The basin, which opens into Boston Harbor and

Massachusetts Bay, is drained by the Charles River. As discussed in Section 2.4 of this report, this drainage to the ocean means that the MITR site is not subject to flooding.

13.2.8.3 Meteorological Disturbances

Tornadoes are very rare in the metropolitan Boston area. In comparison, hurricanes are fairly common, although high sustained wind speeds are rare in this area. The containment building effectively isolates the reactor from meteorological disturbances. The relevant considerations are:

a) The containment building was conservatively designed to conform to the wind load criteria of Massachusetts building codes.

b) The reactor ventilation stack was designed to withstand wind loads in excess of 100 mph.

c) The containment building is protected against excessive pressure variations by the vacuum relief breakers and by the pressure relief system. (See Sections 6.5.4.1 and 6.5.4.2 of this report.)

d) The building and ventilation stack have not sustained any damage from any meteorological condition that has occurred since their construction in 1958. Moreover, both are maintained in excellent condition through regular painting of the building and use of cathodic protection, as well as through periodic inspection of stack.

13.2.8.4 Seismic Event

The MITR is protected against the consequences of a seismic event provided that the core tank remains functional. Other types of damage such as disruption of process systems and lack of electrical power may or may not occur. However, these will not result in core damage for the reasons summarized in Sections 13.2.3, 13.2.4, and 13.2.7 of this report. Specifically, the location of core tank penetrations, the presence of redundant anti-siphon valves, the provisions for natural convection cooling, and the shutdown of the reactor on loss of electricity protect the core against damage to piping and power supplies. Disruption of process systems would, at most, result in the spillage of primary coolant or heavy water. Any such spills would be contained within the reactor building. A seismic event would also not interfere with the capability to shut the MITR down. The six shim blades can not be withdrawn so far that the blades are ever completely out of the blade slots. Thus, blade insertion would not be jeopardized by seismic activity. This is further discussed in Section 13.2.8.6 of this report. The issue therefore is the integrity of the core tank. The impact of a seismic event on the tank is summarized below:

a) Support Structure: The MITR's light-water core tank is supported by the heavy-water reflector tank which in turn rests on the lower annular ring. The lower annular ring is supported by the inner section of the radial thermal shield which is contained within the biological shield That shield is an integral part of the remainder of the reactor containment building. The building, which is 70 feet in diameter, rests on a three-foot thick concrete pad that sits on compacted gravel which is above a layer of "Boston" blue clay that is over a hundred feet deep. (See Section 2.5.2 of this report for a summary of soil conditions.) Because of this construction, it is expected that the shield and pad will shift as a unit rather than crack under seismic shock.

b) Design Acceleration: Analysis of the New England area's geologic history suggests that, for design purposes, a reasonable peak acceleration for a seismic event is 0.15 g. (See Section 2.5.4 of this report.) The MITR is built on Class B soil and such ground tends to amplify seismic motion. Accordingly, a site modification factor of 1.5 is applied. (See Section 2.5.6 of this report.) Hence, the design value applicable to the MITR is 0.225 g.

c) Method of Analysis and Results: Both the light-water core tank and the interior structures (shroud and core support housing) were analyzed to determine the accelerations (horizontal and vertical) that would cause the components to experience their limiting stress. Each structure was divided into a number of elements or nodes for purposes of the calculation. In order to simplify the calculation, it was assumed, as is reasonable in earthquake analysis, that for all cases, the vertical acceleration was two-thirds of the horizontal. It was found that the light-water tank (and not the shroud or core support housing) is the limiting factor. Horizontal and vertical accelerations of 5.1 g and 3.4 g respectively would cause the weakest point in the tank to reach its yield stress of 9500 psia. If one takes a safety factor of fifty percent into account, then a "working" stress limit of 6250 psi is used in lieu of the actual yield stress. This figure is attained for horizontal and vertical accelerations of 2.9 g and 2.0 g respectively. The location at which this limiting stress is attained is the section of the light-water core tank that supports the core support housing. Details of the calculations are given by Allen [13-18].

d) Conclusion: The MITR light-water core tank is capable of withstanding (at yield stresses) static forces corresponding to 5.1 g horizontal acceleration and simultaneously 3.4 g vertically. It is not conceivable, even under the most unfavorable circumstances, that the response to earthquake motions would be more than a small fraction of these amounts.

A seismic analysis was also performed of the MITR's 150-foot ventilation exhaust stack. For this analysis, it was assumed that the seismic event occurred simultaneously with a 100 mph wind. The allowable shear stress was found to be greater than the total (seismic plus wind) for all elevations. Hence, a collapse of the stack would not be expected [13-18].

13.2.8.5 Mechanical Impact or Collision with Building

Mechanical impacts or collisions of some type could occur with either the side or the domed roof of the reactor containment building. The latter is more limiting because the dome consists of 5/8 inch steel plate while the sides are made of 3/8 inch steel plate that encloses two feet of concrete. Accordingly, notwithstanding the design features that preclude such a collapse, an analysis is given here of the effect of a failure of the MITR's exhaust ventilation stack such that the stack falls onto the domed section of the containment.

For purposes of this analysis, the stack was assumed to be hinged at its base and to fall as a unit toward the containment. Figure 13-5 shows the assumed pattern for the collapse of the stack onto the dome. The masses of the upper two portions, which are labeled sections #1 and #2 in the figure, were doubled to account for the effect of impact. Also, the lower section of the stack (height < 39 feet) was not included in the calculation because it can strike only the side of the building. Table 13-5 gives the loads used in the analysis. The maximum total loading is 5.65 psi. The question then arises as to whether the dome will either buckle or fracture.

Table 13-5

Loads Used in Stack Collapse

Zone Section Struck by Mass of Stack	Equivalent Mass of Stack Hitting Section (lbs)	Area Struck (ft^2)	Equivalent Load (psi)
1 and 2	113,700*	160	5.0
3	37,900	90	3.0
4	64,900	104	4.3
5	69,200	112	4.3
6	90,000	119	5.3

* Equivalent mass is increased by a factor of two to simulate dynamic impact effects.

a) Buckling: The critical pressure to cause local buckling of a spherical shell is:

$$P_{cr} = 0.365\, E(t / R)^2 \tag{13-2}$$

where

P_{cr} is the critical buckling,

E is the modulus of elasticity,

t is the shell thickness, and

R is the shell radius.

For the MITR dome, the values of E, t, and R are 0.3 x 10^8 psi, 3/8 inches, and 840 inches, respectively. The critical pressure is therefore 2.3 psi. This is less than the loading calculated in Table 13-5 Hence, local buckling of the containment dome can be expected.

b) Fracture: The dome is made of A-283-C steel for which the yield point and ultimate tensile strength are respectively 33,000 psi and 60,000 - 72,000 psi. The peak stress on the dome, which will occur on the inside surface, was calculated using the same approach as outlined in Section 13.2.8.4 of this report. The peak values are about one-third of the yield point [13-18]. Hence, fracture will not occur. (Note: The figures for the yield point and ultimate tensile strength assume that the steel is above its null ductility transition temperature of -18°C (0°F). The containment building is always kept at room temperature (~22°C). Hence, the temperature of the shell is always above the transition temperature.)

To summarize, local buckling of the domed portion of the containment is likely if the stack should topple onto it. However, fracture will not occur. The twenty-ton polar crane, which is supported by the two-foot thick concrete wall that forms part of the side of the containment, would limit any buckling of the roof. Further protection to the reactor core would be provided by the biological shielding and the reactor top shield lid. (Note: The above analysis assumed a uniform 3/8-inch thick steel shell. The actual thickness is greater over most of the dome.)

13.2.8.6 Seismic Effects on Shim Blades

An analysis of seismic effects on shim blade operation was performed by Allen [13-18]. The principal findings were:

a) Each control blade is connected to a guide rod which traverses a guide tube during vertical movement of the blade. The rods are keyed in the guide tube and thus no rotary motion is possible even during lateral ground movement.

b) Jamming of the guide rod in its tube is not an issue because the length of the guide rod is insufficient to have an angle sufficient to permit jamming.

c) The shim blades are not prone to whip because they are constrained at the top by the guide rod and at the bottom by the blade slots. The displacement of the end of the blade resulting from a 1 g lateral load was calculated to be only 0.16 mm (.0063 inches), which is negligible.

d) Vertical ground displacement could cause a temporary upward force on the shim blades. If this force occurs at the exact moment of blade release and if the blade has an initial upward velocity of 0.1 m/sec (0.3 ft/sec) (which is a reasonable value for a strong earthquake), the blade drop time will only be delayed by 0.02 seconds, which is negligible when compared to a typical blade drop time of 0.7 seconds.

13.2.8.7 Pipe Vibrations

Piping in the reactor was also analyzed to determine if seismic vibrations could cause damage [13-18]. The longest unrestrained run of a major pipe is the light water coolant pipe which runs from the equipment room to the core tank. (The pipe is actually restrained against large motions by the compactness of the area through which it passes.) The fundamental frequency of this pipe is 23 cycles per second, well above normal earthquake frequencies. (Note: Even if a failure of this pipe were to occur, it would not result in core damage because of the anti-siphon valves.)

13.2.8.8 Explosions or Toxic Releases

The containment building would protect the reactor and those utilizing it from both explosions and toxic release.

a) Explosion: For an explosion to have effect, it would have to penetrate both the containment's steel shell and the two-foot thick concrete wall, propagate across the open space within the containment building, then penetrate the 5.5-foot thick biological shield that is made of high density concrete, and then still have sufficient effect to damage the steel thermal shield that is the basis of support for the core tank.

b) Toxic Release: Protection could be obtained against any such release by sealing the containment building. The principal concern is an inadvertent release of liquid ammonia which is used as a refrigerant by local industries. The MITR is equipped with an ammonia detector that senses the air being drawn into building ventilation and which alarms in the reactor control room. The MIT Safety Office, local industries, and the City of Cambridge's Department of Emergency Preparedness exchange information and conduct drills/exercises that involve scenarios such as this.

13.2.9 Mishandling or Malfunction of Equipment

The MITR has been designed to exploit passive safety features. Thus, instrument or equipment failures have been anticipated and generally result in a reactor shutdown. For example, both the safety system and the instruments that monitor radioactive effluents are themselves monitored for power and on-scale indication. If these are lacking, a reactor scram and/or a building isolation occurs. However, there are some situations for which it has not been possible to devise interlocks and/or a fail-safe response. For these, administrative procedures are observed. These situations are discussed here.

13.2.9.1 Operation with Shim Blades in a Non-Uniform Bank Position

It is intended that, whenever the MITR is at a power level in excess of 1 kW, the six shim blades be within ±2 inches of an average bank height. This is done in order to achieve a uniform distribution of the neutron flux. An interlock is used during reactor startups to assist the console operator in achieving this objective. Specifically, the subcritical interlock, which is described in Section 7.3.1.2 of this report, blocks blade withdrawal beyond four inches unless all blades are first brought to the four inch position. Once all blades are above four inches, satisfaction of the requirement to maintain a uniform bank height is achieved by administrative procedure. Hence, either operator error or an incorrect indication of blade position could result in a non-uniform bank. This would in turn cause a tilt in the power distribution.

Calculations have been made of the effect of operating with one blade full in and the other five at a shim bank height of ten inches. These were done using the computer code CITATION [13-19]. These calculations showed an increase in the power density on the side of the core opposite to the inserted blade of about 4%. Hence, the effect of operation with one blade fully inserted would be to increase the radial peaking factor for the hot channel by a factor of 1.04 (worst case). Such an increase will not cause the hot channel to approach the safety limit. For the MITR at its LSSS of 7.4 MW, the hot channel would be at 44.8 kW which is well below the hot channel power of 67.9 kW that corresponds to the safety limit based on a hot channel factor of 2.

13.2.9.2 Use of Lead Fixtures Over the Core Tank

The MITR is a tank-type reactor. Storage for spent fuel is provided by a pool that is separate from the tank. That is, there is no canal that connects the two. In order to transfer spent fuel from the core to the pool, a lead transfer cask is used. The procedure is done with the reactor shut down. The reactor top shield lid is removed and the element in question, which is normally already in the fuel storage ring, is placed in a protective basket. The lid is replaced and the plug to one of the ports that penetrates the lid removed. A lead sleeve is lowered through the port and supported on the port's lip. The sleeve provides shielding. The fuel transfer cask, which is a lead-filled annulus, is then placed over the port so it rests on the lid. The basket is drawn into the cask, the cask shutter is closed, and the cask transferred to the spent fuel pool. The various lead fixtures are moved using the overhead polar crane. Certain administrative requirements are imposed on the above procedure:

a) The lead fixtures are not moved across the top of the core tank unless the reactor top shield lid is in place.

b) The bottom of the fuel transfer cask is kept within 6 inches of the upper surface of the reactor top shield lid.

If these requirements are met, analysis has shown that no damage will occur to the core should one of the lead fixtures and/or the cask drop onto the lid. The lid would deform but it would not rupture [13-20].

13.2.9.3 Spill of Heavy Water

Leakage from the heavy water reflector is of concern because of the tritium contained in the heavy water. The heavy water system is equipped with a leak detection system that will sound an alarm to notify the operator of any moisture that may result from a leak. This system is documented in Section 5.3.1.10 of this report.

The maximum allowable tritium content in the heavy water is 5 Ci/liter as indicated in Section 5.3.1.6 of this report. An evaporation rate of 5 liters per hour is estimated for a heavy water temperature of 55°C, a relative humidity of 20%, and a spill area of 6 feet in diameter. This evaporation rate corresponds to a total volume of 120 liters, or a maximum activity of 600 Curies, in 24 hours. This is higher than would actually occur and hence conservative.

The effluent concentration (EC) for tritium concentrations in air in unrestricted areas is 1×10^{-7} μCi/ml. Using this value, a conservative dilution factor of 900, and assuming the containment building ventilation is operating, it is calculated that approximately 20 EC / 3 mrem exposure off-site would be attained over 24 hours [13-21].

13.2.9.4 Mixing of Light and Heavy Water

See Section 4.5.1.6 of this report.

13.3 Summary and Conclusions

This chapter contains conservative analyses of different types of accidents that are related to the MITR and its surrounding environment. There is no projected damage to the reactor core as an outcome of the accidents evaluated, except when the core damage is assumed to be part of the accident scenario, as described in Section 13.2.1 of this report, the maximum hypothetical accident. No radiation exposure to the public is expected except in the maximum hypothetical accident, for which the maximum whole body dose to the general public is 381 mrem during a two hour period at 21 m from the containment building.

The robustness of the MITR is the result of the use of passive and engineered safety features in its design. These features are enumerated in Section 1.2.3 of this report.

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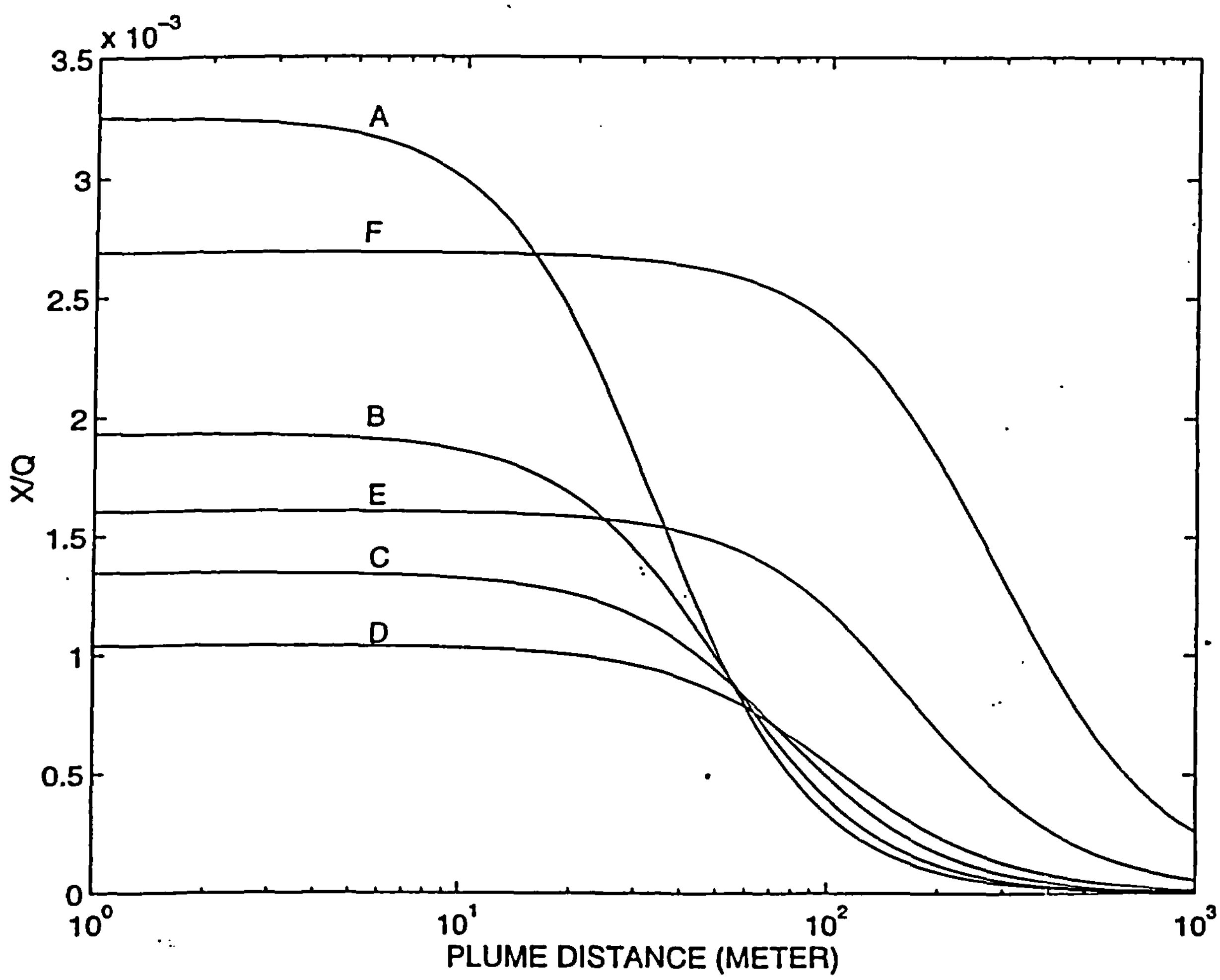


Figure 13-1 Relative Atomspheric Concentrations (X/Q) as a Function of Plume Distance for Each Atmospheric Condition from Containment Leakage using the " Exact" Model. [13-4]

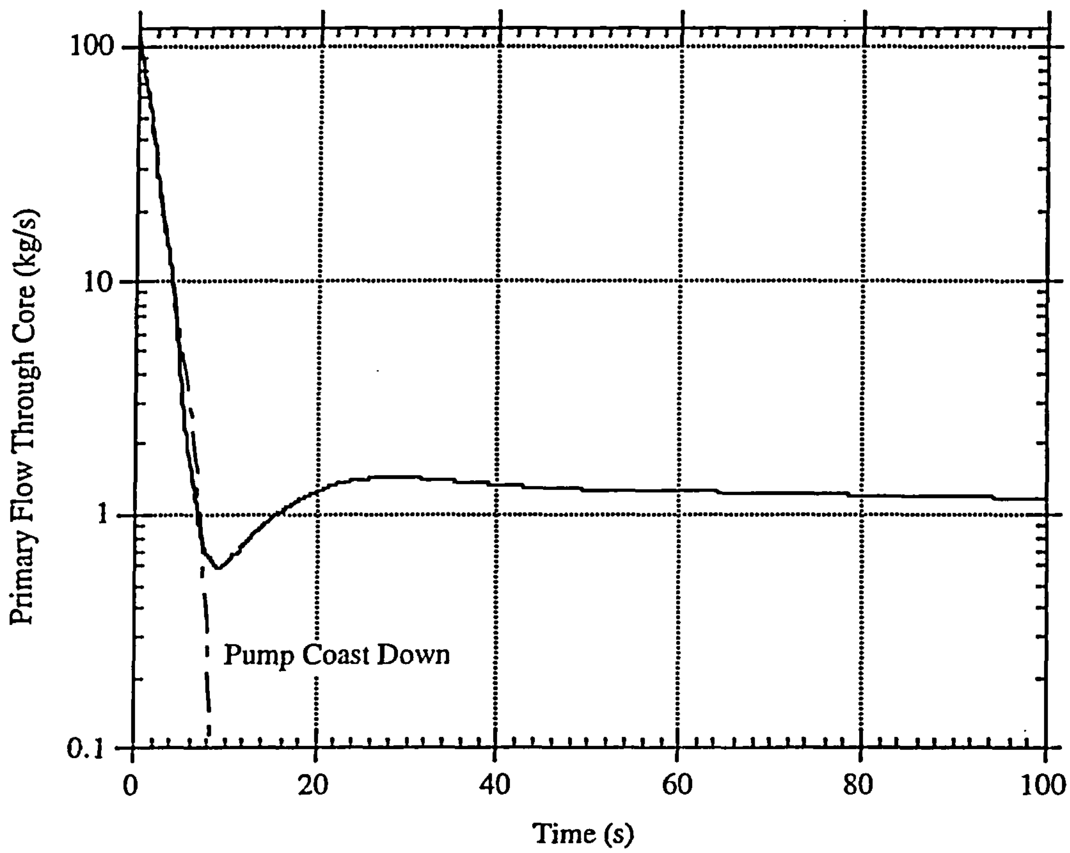


Figure 13-2 Primary Flow through the MITR Core During a Pump Coast-Down Transient.

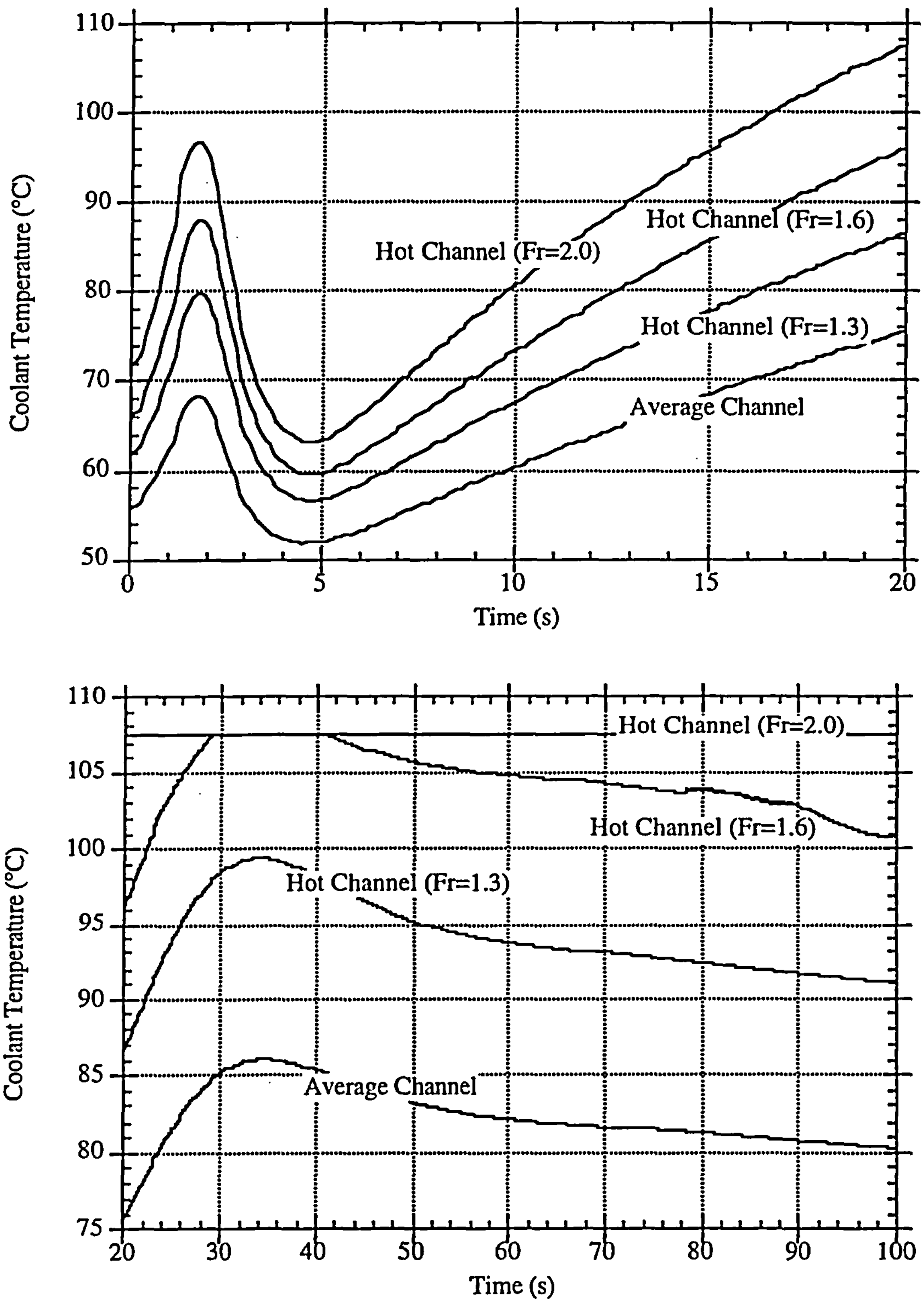
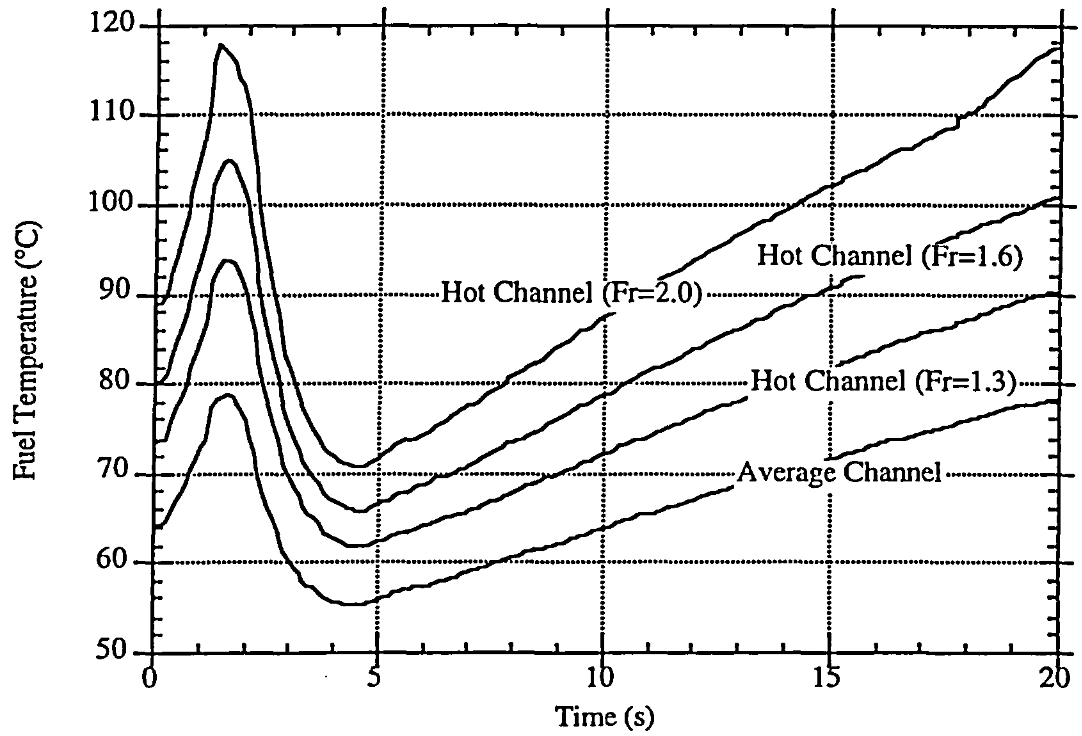


Figure 13-3 Coolant Outlet Temperatures for the Hot and Average Channels During a Pump Coast-Down Transient.



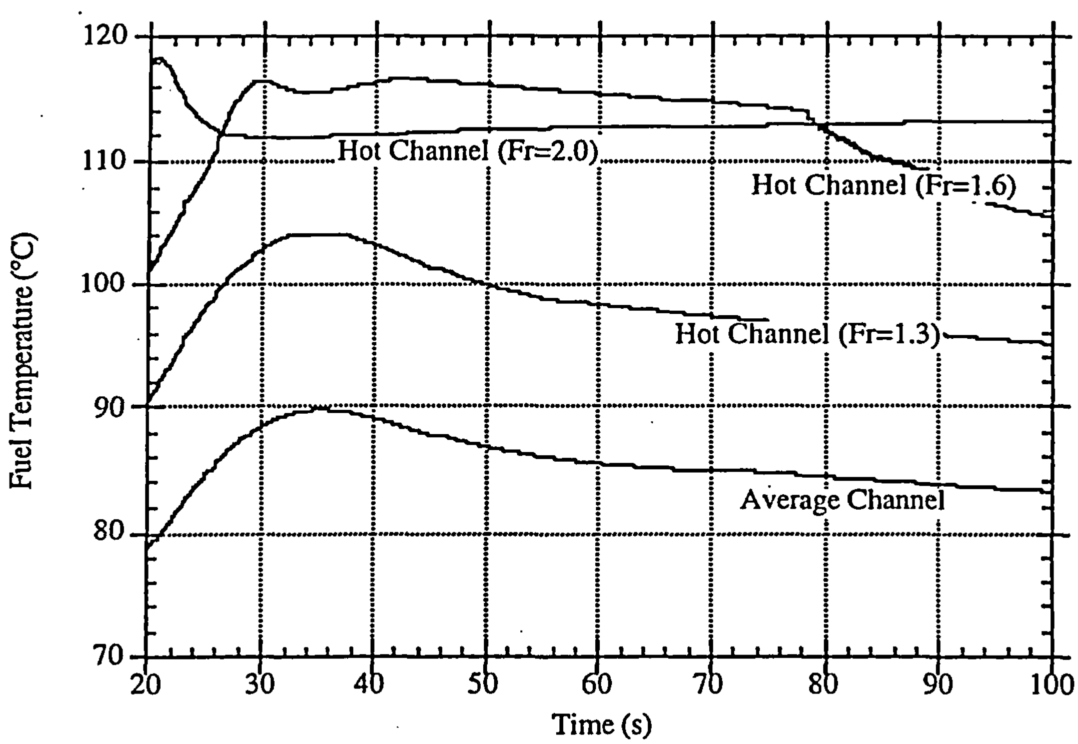


Figure 13-4 Fuel Temperatures at the Channel Outlet for the Hot and Average Channels During a Pump Coast-Down Transient.

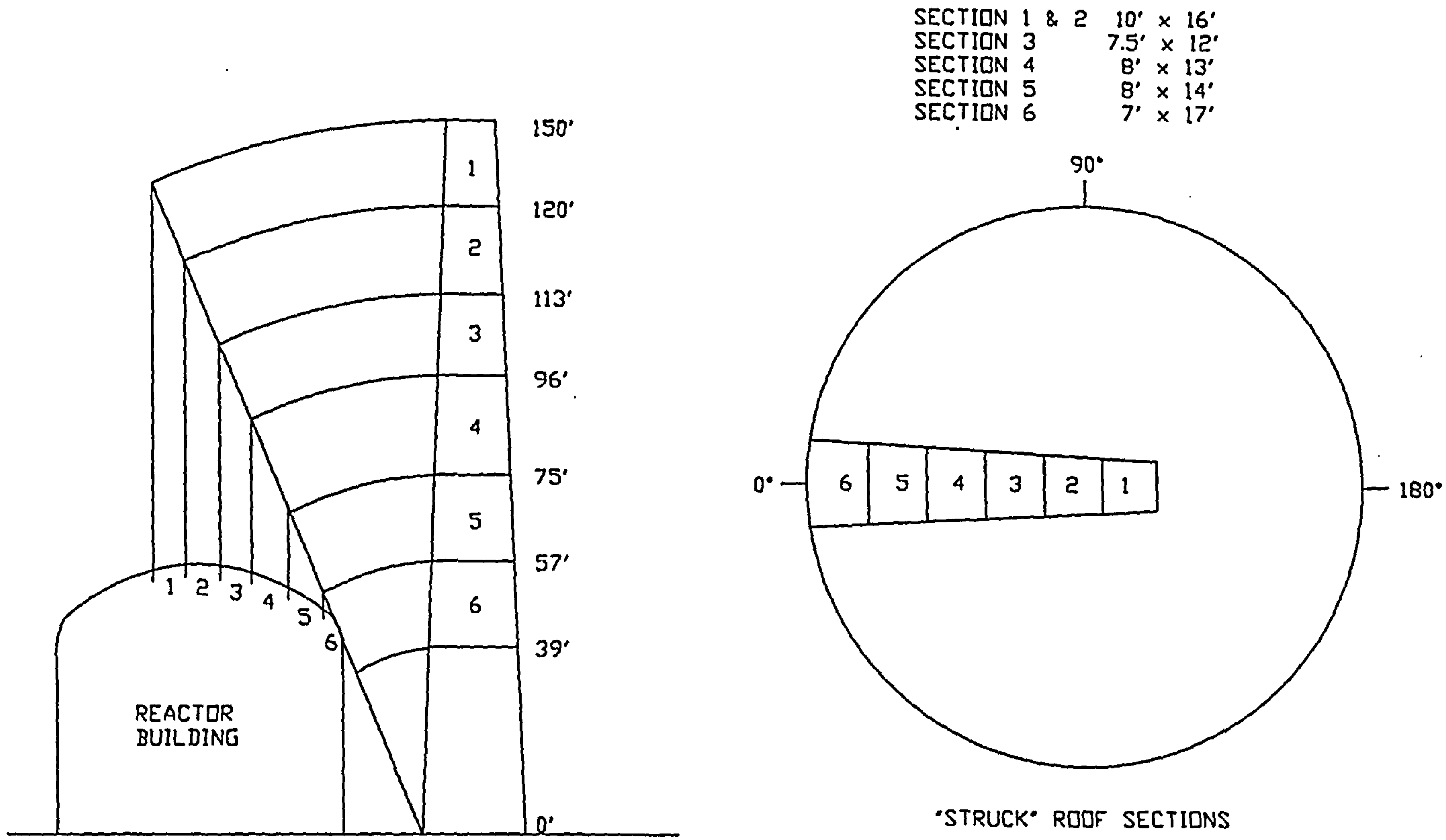


FIGURE 13-5
MODEL OF STACK COLLAPSE

Appendix A

Synopsis of Boiling Coolant Chemistry Loop (BCCL) Safety Evaluation Report

A.1 Facility Purpose

The BCCL is used to carry out research into the effects of radiolysis and chemical additions (hydrogen, for example) on coolant composition (oxygen and hydrogen peroxide concentration, for example), as part of a program to evaluate the effect of the coolant environment on the corrosion of materials in Boiling Water Reactors.

A.2 Facility Description

Figure A-1 is a schematic of the major features of the BWR Coolant Chemistry Loop (BCCL). It is a once-through system: high-purity feedwater is pumped from out-of-pile through the in-pile Zircaloy core section, where approximately 10 weight percent boils. The effluent is then separated in an outlet plenum and taken out-of-pile for analysis, and ultimate re-use. The overall layout is shown in Figure A-2. From the right: cold feedwater is pumped by a charging pump rated at ≥1200 psi at a rate of approximately three liters per minute through a regenerative heat exchanger, and an electrically-heated feedwater heater to the in-pile thimble. The same total mass of water, as steam and liquid at ≈300° C, is then brought back out of the MITR core tank, where the two effluent streams are combined, lose heat in the regenerative and non-regenerative heat exchangers, and are returned to the hotwell tank.

The containment thimble is the actual interface between the MITR core and coolant and the subject experiment. The material of construction (aluminum) and all shapes/dimensions are the same as thimbles used in previous ICSAs. The section of the loop that is above the fueled region of the MITR core is beveled at its bottom, and positioned such that "shadowing" of adjacent core fuel assemblies, from the emergency cooling spray, is minimized. Thus no interference in either normal core coolant flow or emergency core spray reflood should take place.

Table A-1 lists important BCCL design and operating parameters [A-1].

Table A-1

Important BCCL Design and Operating Parameters

Outlet Plenum Volume (water and steam)		1200 cc
Plenum Pressure and Temperature		1000 psia, 545° F
Downcomer Plenum Volume		55 cc
Feedwater Flow Rate		3000 cc/minute
Core Section (Zircaloy U-Tube) Inlet Temperature		530° F
Length of In-core U-tube		48 inches
Maximum Core Effluent Steam Quality		15 w/o
In-Pile Electric Heater Rating		20 kW
Feedwater Electric Heater Rating		20 kW
Maximum H_2 Content in Feedwater		2 cc/kg
In-core Dose Rates	Neutron	4×10^8 R/hr = ≈1.6 w/g
(Mean, in H_2O)	Gamma	4×10^8 R/hr = ≈1.6 w/g
Operating Temperature of Hotwell Tank (max)		100° F
Rating of Heat Exchanger	Regenerative	36 kW
	Non-Regenerative	7.5 kW
Void/Reflood Reactivity of In-Core U-Tube		$0.042\% \frac{\Delta K}{K} = 53.4$ mbeta

A.3 Compliance with MITR-II Technical Specifications

A.3.1 Technical Specification Design Criteria

MITR-II Technical Specification #5.2.2 specifies that in-core experiments must comply with four criteria. These are:

a) They shall be positively secured in the core to prevent movement during reactor operation.

b) Materials of construction shall be radiation resistant and compatible with those used in the reactor core and primary coolant system.

c) Sufficient cooling shall be provided to ensure structural integrity of the assembly and to preclude any boiling of the primary coolant.

d) The size of the irradiation thimble shall be less than sixteen square inches in cross section.

The proposed BCCL design meets these criteria as noted below:

A.3.1.1 Lack of Movement During Operation

The BCCL, in the same manner as all previous in-core sample assemblies, is secured via the upper grid plate.

A.3.1.2 Materials of Construction

The exterior of the BCCL is aluminum and hence is compatible with the MITR-II's fuel, core vessel, and water chemistry.

A.3.1.3 Heat Removal

Energy is generated by means of a 20 kW electric heater and a lead bath (gamma-ray attenuation). Energy removal is achieved through use of a copper shot bed in the riser section of the loop. The BCCL rejects heat outside the reactor tank. The BCCL is protected by redundant sensors that will shut off the electric heater if the temperature becomes excessive. Should that cutoff fail, an alarm will sound in the control room and the operator can de-energize the system. This arrangement satisfies Technical Specification #5.2.2(c)

A.3.1.4 Cross-Sectional Area

The BCCL occupies only one fuel element position. Its cross-sectional area is less than sixteen square inches.

A.3.2 Reactivity of the BCCL Experiment

MITR-II Technical Specification #6.1.1 limits the reactivity worth of experiments to the following values:

	Single Experiment Worth	Total Worth
Movable	0.2% $\Delta K/K$	0.5% $\Delta K/K$
Non-Secured	0.5% $\Delta K/K$	1.0% $\Delta K/K$
Total of the above	---	1.5% $\Delta K/K$
Secured	1.8% $\Delta K/K$	---

The three types of experiments are defined in Section 1 of the Technical Specifications as follows:

a) Secured Experiment

A secured experiment is an experiment or experimental facility held firmly in place by a mechanical device or by gravity, such that the restraining forces are substantially greater than those to which the experiment might be subjected by hydraulic, pneumatic, buoyant, or other forces which are normal to the operating environment of the experiment or by forces which can arise as a result of credible malfunctions.

b) Movable Experiment

A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into or out of the reactor while the reactor is operating.

c) Non-Secured Experiments

Experiments where is intended that the experiment should not move while the reactor is operating, but is held in place with less restraint than secured experiment.

A.3.2.1 BCCL Water and Contained Chemicals

For the BCCL, a change of phase is intended. Hence, the water and contained chemicals are classed as "movable." The limit is therefore 0.2% ΔK/K or 254 mbeta. Estimates put the reactivity worth of the coolant at +54 mbeta.

A.3.2.2 Flooding/Reflooding

Flooding/reflooding scenarios were examined for the accessible volume of the BCCL. This includes the void volume in the thimble and the coolant channel annulus between the thimble and the dummy element. The effect of flooding those spaces is classed as "non-secured" because it is not intended that these spaces be flooded or voided during normal operation. The estimated reactivity is +181 mbeta. Hence, the flooding/reflooding scenarios meet the "non-secured" limit which corresponds to 0.5% ΔK/K or 636 mbeta.

A.3.2.3 BCCL Components

The BCCL components (lead, loop, heater, and fixtures) are classified as a secured experiment, because they are mechanically held in position by the loop tubing and other structural components. Their complete ejection from the thimble followed by flooding must not exceed 1.8% ΔK/K. The reactivity worth is -407 mbeta. The absolute value of this figure is well below the limit of 1.8% ΔK/K or 2290 mbeta.

A.3.2.4 Experimental Interactions

Technical Specification #6.1.1 imposes limits on the total amounts of reactivity associated with both movable and non-secured experiments. Initially, the BCCL will be the only in-core experiment. However, at some future time, two loop experiments might be run simultaneously. It is noted here that the MITR-II core could contain three such experiments and not exceed either the movable or the non-secured limit.

A.3.3 Pressure Effects

Technical Specification #6.1 requires that experiments be designed to withstand twice the anticipated pressure. The loop portion of the BCCL will operate at approximately 1000 psi. Surrounding the actual loop is an elliptical thimble made of 6061 aluminum. It would be pressurized in the event of failure of the BCCL internals. The thimble is protected by a pressure relief valve set at 15 psi and a burst disk set of 65 psi. The maximum expected pressure is therefore 65 psi. The thimble will therefore be hydrostatically tested to 150 psi.

A.3.4 In-Core Boiling

The BCCL experiment will involve boiling of the loop coolant which will, of course, be physically within the core. The MITR-II's technical specifications do not prohibit boiling within an experiment (as opposed to boiling of the primary coolant itself). The relevant technical specifications are:

A.3.4.1 Technical Specification #2.2 - Limiting Safety System Settings

The objective of this specification is: "to assure that automatic protective action will prevent incipient boiling in the reactor core and will prevent conditions from exceeding a safety limit." The specification itself establishes limiting conditions for flow, power, core tank level, and temperature to" ...prevent incipient boiling... which is initiated prior to the initiation of flow instability."

As to the applicability of this specification, the specification section itself contains no prohibition against boiling in an experimental facility. Its purpose is to restrict certain reactor parameters so that there will be no problem with flow in the fuel element channels. Accordingly, it is concluded that this specification does not pertain to in-core experiments that are isolated from the core, such as the BCCL.

A.3.4.2 Technical Specification #5.2 - Reactor Core

Paragraph 2(a) of this specification states that the "Design of in-core sample assemblies shall conform to the following criteria," one of which is that "sufficient cooling shall be provided to ensure structural integrity of the assembly and to preclude any boiling of the primary coolant. This specification clearly requires that the experiment be designed so that there will be no boiling on the outer surface of the experiment thimble and tube. The type of boiling, nucleate or bulk, is not mentioned. Presumably both are prohibited. There is no prohibition of boiling within the thimble.

A.3.4.3 Technical Specification #6.1 - General Experiment Criteria

Paragraph 2(b) of the specification states that, "The outside surface temperature of a submerged experiment or capsule shall not cause nucleate boiling of the reactor coolant during operation of the reactor." This specification clearly requires that the experiment be designed so that there will be no nucleate boiling of the coolant. There is no prohibition of boiling within the thimble.

A.3.5 Explosion Hazard

Technical Specification #6.1.3(b) limits explosive materials to the equivalent of 25 mg of TNT. The BCCL will have a hydrogen gas concentration of 2 cc/kg. Assuming this concentration is in all of the BCCL coolant that is within the reactor's biological shield, the total amount of hydrogen present will be about 2.5 cc at standard temperature and pressure. This corresponds to less than a milligram equivalent of TNT.

A.4 Safety Analysis

In the previous section of this safety review, the compliance of the BCCL experiment with the MITR-II's technical specifications was shown. In this section, specific safety issues are examined with the objective of demonstrating that an "unreviewed safety question" or URSQ, does not exist. For the record, an URSQ is defined in 10 CFR 50.59(2) to exist if:

> A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously valued in the safety analysis report may be increased; or (ii) if a possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or (iii) if the margin of safety as defined in the basis for any technical specification is reduced.

A.4.1 Temperature Effects

The in-pile loop assembly will be heated both by a 0-20 kW heater and by a combination of gamma and fast neutron radiation. The normal combined heat load will be less than 20 kW. The radiation heating is estimated to be 9.6 kW at 5 MW [2]. Hence, 29.6 kW would be the maximum heat load potentially available under malfunction conditions. This is not much more than the hottest running fuel plate in the MIT Reactor. Hence, the thimble is easily cooled by the flow of primary coolant through the 0.050 inch thick channel between the thimble and the dummy fuel element that surrounds it.

The potential for a Zircaloy-water reaction was analyzed in the PCCL SER [2] where it was shown that cooling by conduction and radiation should prevent temperatures in the thimble from exceeding 1845° F for the maximum radiation heating, i.e., 9.6 kW. This was based on a very conservative extrapolation of temperatures measured in a test mock-up of the PCCL assembly operating out-of-core at heater powers in the range of 2470-4510 watts. The 1845° F temperature is significantly below the 2200° F post-LOCA limit on Zircaloy temperature imposed on PWR units by the U.S. Nuclear Regulatory Commission [3].

Assurance that total heating will not exceed the 9.6 kW which might result from gamma and fast neutron heating with the reactor at full power is achieved by redundant heater shut-offs that are activated by high lead bath temperatures. The sensors and relays that interrupt power to the heaters are completely independent, thus avoiding compromise by a single failure.

Elevated lead bath temperatures are not a threat to the aluminum thimble, because there is no contact between the thimble and the titanium can that holds the lead except at occasional small points of contact with high spots on the weld bead stiffener on the outer surface of the titanium can and at the support ring which is at the top of the titanium can extension about 12 inch above the lead bath.

In view of the above active and passive safety features, it is not credible that temperature effects within the thimble can affect the fuel, core structure, or other components important to safety and, hence, there is no unreviewed safety question in this regard.

A.4.2 Hydrogen Leak and Combustion

This issue was thoroughly reviewed as part of the PCCL SER [4]. The concentration of hydrogen gas in the PCCL coolant is approximately 20 cc/kg. That in the BCCL's coolant will be 2 cc/kg, a factor of ten less. Hence, the BCCL's use of hydrogen is well within the envelope of previously established conditions.

The hydrogen gas used for the BCCL will be stored and handled using the same procedure as is now followed for the PCCL. The only difference is that the total inventory of hydrogen gas present in the containment will be doubled. Specifically, a storage limit of 20 SCF per loop not to exceed a total of 80 SCF will be observed. This does not present a hazard because discharge of 80 cubic feet of hydrogen into the containment building volume of 200,000 ft^3 will result in a concentration far below the lower explosive limit of 4.1%. Moreover, there is a fan mounted near the hydrogen station that ensures rapid dispersion of the gas. The maximum amount of hydrogen gas stored in any one container will not exceed 20 SCF.

A.4.3 Loss of Loop Pumping Power or Loss of Flow

The heat sources for the BCCL are a 0-20 kW electric heater and a lead bath for gamma ray attenuation. The heat sink is a non-regenerative heat exchanger located outside of the reactor vessel.

In the absence of any human interaction following a loss of BCCL flow, the in-core lead heater bath temperature will increase until the redundant over-temperature trips cut off electric heater power, following which gamma heat will be safely rejected by passive means, radiation and conduction to the thimble wall, and thus to the MITR-II coolant.

A.4.4 In-Thimble Leakage

Section 2.A.3 of the BCCL SER that was provided by the experimenter addressed the issue of whether a rapid, large leak could result in an overpressurization of the thimble. The MITR Staff agrees with the analysis given and with the conclusion that the relief valve and burst disks installed on the thimble are adequate. (Note: Section 2.A.3 is not reproduced in this synopsis.)

A.4.5 Lead Bath Can Leak

Section 3.6 of SR #0-86-9 (PCCL Safety Review [4] addressed the issue of large and small leaks of lead from the titanium can and concluded that there was no credible mechanism by which the loop could adversely affect MITR safety. That conclusion applies here because the can construction is identical.

A.4.6 Electrical System Malfunction

The electric power supply for the BCCL is essentially identical to that for the PCCL. Section 3.8 of SR #0-86-9 (PCCL Safety Review [4]) examined the possibility of an electrical short and concluded that the loop was properly designed to prevent such an occurrence. The same analysis applies to the BCCL and is therefore not repeated here.

A.4.7 Effect of Boiling on Reactor Operation

The concern here was that the presence of boiling in the BCCL might cause reactivity fluctuations which, while within the magnitude allowed by the technical specifications, would cause operational problems such as excess movement of the regulating rod. An experiment was conducted to resolve this issue. Known as the "In-Pile Boiling Experiment," it involved performing boiling tests in the PCCL while at low power. Safety aspects of the proposed experiment were documented in SR #0-88-6 and approval for the experiment was given by the Special Subcommittee of the MITRSC on 09/15/88 and by the full MITRSC on 12/20/88.

The "In-Pile Boiling Experiment" was conducted on 06/19/89 and it was found that the presence of boiling in the PCCL in-core section did not have any discernible effect on MITR-II operation. There is no credible mechanism by which the BCCL can adversely affect MITR safety on this issue.

A.4.8 Emergency Core Cooling System (ECCS)

The concern here was that the BCCL might shadow some of the fuel elements from the water sprayed onto the top of the core by the ECCS system in the event that the reactor core should not be covered by water. This shadowing could occur either as the result of the BCCL's riser section or as the result of the gamma ray pod.

Tests were made using a mockup of the core top, primary coolant flow guide, and the BCCL. It was found that sprayed ECCS water splashed randomly from the experimental facilities and from the interior surfaces of the flow guide so that any shadowing effect was minimized, and each fuel position received at least 40% of the average flow per element. Five tests were performed and the results averaged. The tests were conducted according to PM 6.1.1 which is used to verify compliance of the installed ECCS system with the MITR Technical Specification criteria on an annual basis. Technical Specification #3.6 requires that a total of 10 gpm be delivered via the ECCS system. Section 6.1 of the MITR-II's SAR further requires that each fuel element receive at least 20% of the average flow per element. All fueled element positions received at least 42% of the average, which is well above the 20% required. Accordingly, the installation of the BCCL will not interfere with the proper functioning of the emergency core cooling system. (Note: The mock-up tests were performed for the BCCL alone. Should the PCCL and BCCL be run simultaneously, a new set of tests would be required.)

A.5 Conclusion

It is concluded that failures or accidents originating with the BCCL loop cannot interact with the reactor fuel, core structure, or other components important to safety, except through reactivity effects. In this case, loop failures or accidents will not cause reactivity changes exceeding those authorized by the MITR Technical Specifications. For equipment important to safety, a) the probability of an accident or malfunction is not increased, b) the possibility for an accident or malfunction of a different type than that previously evaluated in the SAR is not created, and c) no margin of safety in any technical specification is reduced. Consequently, the BCCL experiment does not involve an unreviewed safety question.

REFERENCES

A-1. Safety Evaluation Report of the BWR Coolant Chemistry Loop (BCCL), Report No. MITNRL-031, March 9, 1989

A-2. Safety Evaluation Report for the PWR Coolant Chemistry Loop (PCCL), Report No. MITRNRL-020, February 13, 1987 plus supplement dated March 22, 1988.

A-3. 10 CFR 50.46(b)(1)

A-4. SR #0-86-9, "PWR Coolant Chemistry Loop, April 21, 1988.

Note: The above is a synopsis of the BCCL SER. Also, it does not include startup test results which confirmed the reactivity figures, operating procedures, emergency procedures, activation analysis, etc.

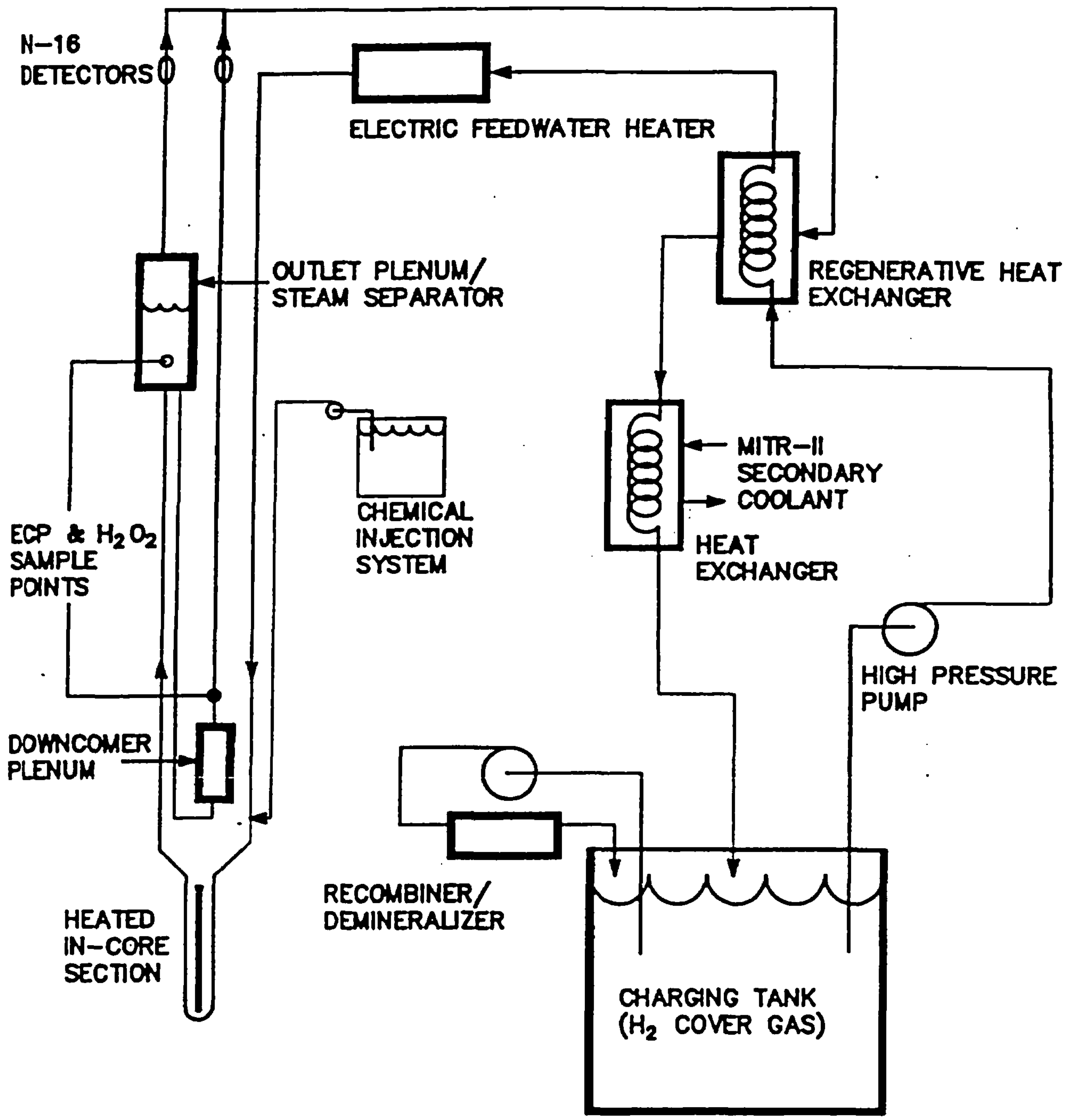


FIGURE A-1 SCHEMATIC OF IN-PILE BWR COOLANT CHEMISTRY SIMULATION LOOP

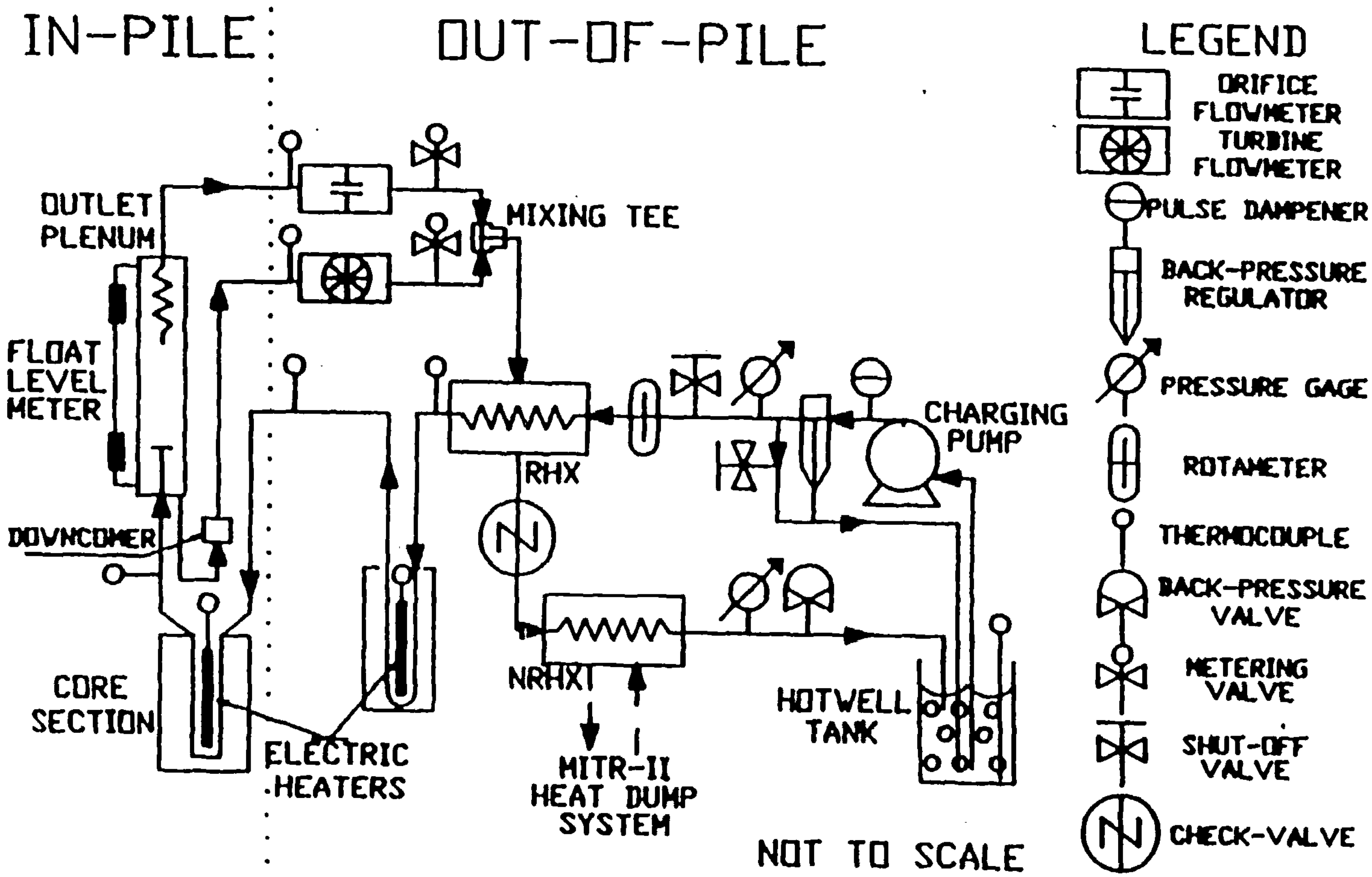


FIGURE A-2 SCHEMATIC OF BCCL CONFIGURATION: OVERALL LAYOUT

Chapter 14

Technical Specifications

14.1 Format and Content

Technical specifications for the MITR are contained in a separate document, MITR Technical Specifications. These specifications were developed in accordance with ANSI/ANS-15.1-1990. Normal operation of the MITR within the limits of these technical specifications will not result in offsite radiation exposure in excess of 10 CFR Part 20 guidelines. Also, observance of these technical specifications limits the likelihood and consequences of malfunctions.

Chapter 15

Financial Qualifications

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Chapter 15

Financial Qualifications

15.1 Summary

The Massachusetts Institute of Technology is financially qualified to own, construct, operate, and decommission a non-power reactor. Given that the MITR is an existing facility, the issue of construction is not relevant here. Accordingly, this Chapter focuses on MIT's financial ability both to operate the facility safely and to decommission the facility.

15.2 Fuel Cycle Costs

The United States Department of Energy (DOE) provides financial support for the MITR fuel cycle. This support includes fuel fabrication, transport of fuel to MIT, and removal of spent fuel from MIT. A copy of the DOE-MIT contract concerning the MITR's fuel cycle is available through MIT's Office of Sponsored Programs.

15.3 Financial Ability to Operate a Non-Power Reactor

The Massachusetts Institute of Technology has the financial ability to operate the MIT Research Reactor. Table 15-1 lists actual operating expenses for the fiscal years 1994-1998. Expenses are broken down by category of spending including salaries, benefits, supplies, and equipment. Overhead is not charged on reactor operating expenses because it applies only to externally-funded accounts such as those associated with research contracts and/or service income.

Certain services are provided at no cost to the reactor operating account. Specifically MIT's Department of Facilities provides cleaning, utilities, and building maintenance and the MIT Medical Department provides health physics and industrial hygiene coverage. These services are paid for with funds produced by overhead charges which are in turn imposed on all MIT accounts that generate research or service income. This approach has certain benefits in terms of safety. Services, such as health physics, are provided to all MIT Departments and Laboratories according to need and not ability to pay.

Table 15-1

Reactor Operating Expenses

Category	FY 94	FY 95	FY 96	FY 97	FY 98
Salary/Wages	836,326	922,484	962,394	985,549	1,227,869
Benefits	325,331	347,776	375,334	401,118	499,743
Materials & Supplies	302,269	350,025	286,348	214,512	212,976
Equipment	168,217	85,927	74,962	97,131	65,046
Travel	16,645	23,567	20,668	23,719	29,390
Other*	–	11,425	32,295	54,296	77,779
Main Frame Computer	1,212	3,161	4,524	3,497	18,399
Total	1,650,000	1,744,365	1,756,525	1,779,822	2,131,202

*Reflects federally-mandated changes in the billing of tuition for student research assistants.

Another category of expense that is not included in the figures shown in Table 15-1 are administrative expenses. Salaries for the Nuclear Reactor Laboratory Director, the Financial Officer, and their secretarial support are provided by MIT from central funds.

The figures shown in Table 15-1 include both the actual cost of operating the MITR and expenses for utilization. The latter average 10% of the total cost.

Income to offset reactor operating expenses is generated by charging users of the reactor for neutrons. For example, an MIT researcher may wish to identify and quantify sources of air pollutants through use of neutron activation analysis. The researcher would obtain funding from a consortium of industrial users and/or a government agency. That funding would then be used to establish an account that would pay the stipends of graduate students who perform thesis research on the project, the salaries of any technical support people, the cost of supplies for the research, and the cost of use of the reactor for sample activation. In addition, the account would bear full overhead.

Charges for use of the reactor are the sole source of income. MIT is a private university and receives no federal or state tax dollars for its base operating budget. Income raised through charges for neutrons has averaged 67% of the operating expenses for the last five years. The remaining 33%, which amounts to several hundred thousand dollars per year, is the operating deficit. It is covered by MIT general funds.

Sources of income vary significantly with time. This is to be expected because research reactors are a means of generating neutrons for use in probing matter. A useful analogy is that of a light microscope that uses photons to reveal structures at the cellular level. A research reactor is also an instrument, albeit a large one, that produces neutrons for the study of matter at the nuclear level. Research interests and hence income sources change with each new application of neutrons. Major sources of income presently include nuclear medicine, especially neutron capture therapy and radiation synovectomy; neutron activation analysis in the fields of geology, nutrition, and pollution control; nuclear engineering with emphasis on materials studies; and silicon transmutation doping.

15.3.1 Commercial Activity

MIT is a non-profit educational institution. The MIT Research Reactor therefore does not engage in commercial work if that work would compete with private U.S. industry. At present, there are no U.S. vendors who perform transmutation doping of silicon for the manufacture of high performance switches for electrical equipment. Accordingly, the MITR (and also a number of other university research reactors) irradiates silicon ingots for various international vendors. The income generated is used to offset operating expenses and thereby support research. Other benefits to this activity are that it maintains a U.S. technological capability that would otherwise be lost to other nations and it contributes to a positive balance-of-payments.

The cost of this commercial activity is modest because the irradiation facility is automated. A maximum of 1.5 effective full time people are required to support this activity. This equates to less than 10% of the reactor operating budget.

15.4 Financial Ability to Decommission the Facility

The Massachusetts Institute of Technology has no plans to decommission the MIT Research Reactor at this time. However, MIT does have the funds necessary to do so.

15.4.1 Estimate of Decommissioning Costs

In 1988 the Massachusetts Institute of Technology contracted with General Electric Company (Decommissioning Projects) to provide a Conceptual Decommissioning Plan and Cost Estimate for the MIT Research Reactor [15-1]. The purpose of this Conceptual Decommissioning Plan and Cost Estimate was to provide the basis for the owner/operator (MIT) to make provisions, in advance, for funding the dismantling of the reactor facility at the end of its useful life. Such funding provisions are required by the Code of Federal Regulations 10 CFR Part 50 "Domestic Licensing of Production and Utilization Facilities."

The scope of work covered removal of the reactor and reactor-associated equipment and material which is required to enable the unrestricted free release of the remaining facilities, where continuous occupancy will result in no more than 10 millirem/year exposure. Removal of the reactor fuel, the control blades, the reactor building, the ventilation stack, and the cooling tower was not considered part of this Conceptual Decommissioning Plan. (Note: Removal of fuel would be paid for by the U.S. Department of Energy.)

The cost of decommissioning was estimated in 1988 to be \$9.1 M. General Electric provided a cost adjustment formula to allow for updating of this estimate. That formula is:

$$\text{Cost} = (1988\ \text{Base Cost})(0.89L + 0.11B)$$

where L is the labor cost and B is the waste disposal cost relative to the 1988 costs. The former can be obtained from U.S. Department of Labor statistics. The latter is a function of the availability of low level waste disposal technologies. The cost estimate for decommissioning the MITR rose dramatically in the early 1990s because of the projected closure of current low level waste sites. In 1996, the cost estimate peaked at \$21.55 M. Since then, the figure has held constant or possibly declined slightly because of stabilization of low level waste cost projections and added experience with labor requirements for decommissioning.

MIT currently maintains an escrow account to cover the cost of decommissioning. It is anticipated that MIT will change to the self-guarantee method in the near future. The U.S. Nuclear Regulatory Commission will be notified of any such change.

References

15-1 "Conceptual Decommissioning Plan and Cost Estimates for the MIT Research Reactor," prepared by GE Nuclear Energy, December 1988.

Chapter 16

Other License Considerations: Prior Use of Reactor Components

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Chapter 16

Other License Considerations: Prior Use of Reactor Components

16.1 Summary of Material Condition

The material condition of all MITR systems important to safety is excellent. This includes the containment building, the core tank and housing, the primary and heavy-water systems, the nuclear and process safety systems, and the area and effluent radiation monitoring equipment. There are four reasons for this excellent material condition. First, several major systems have been recently replaced with new, state-of-the-art equipment. This includes but is not limited to the area and effluent radiation monitoring systems and the cooling tower. Second, certain of the reactor's major components are replaced at regular intervals. This includes the fuel and the reactivity control system's absorbers. Third, some major components were replaced in 1974 as part of the reactor modification that is described below in Section 16.2 of this report. Hence, these components do not date from the issuance of the original MITR license in 1958. Examples include the heavy-water reflector tank, the light-water core tank, the core housing, and the upper and lower fuel element grid plates. Fourth, components that have remained in continuous use have been subject (as are all components) to regular inspections for evidence of degradation and, if any was found, the component was repaired.

16.2 History of the MITR

Table 16-1 is a chronology of the MITR (Operating License No. R-37). Major milestones in the facility's history are:

a) On May 7, 1956, the United States Atomic Energy Commission (AEC) issued Construction Permit No. CPRR-5 to the Massachusetts Institute of Technology (MIT). Construction then began on the original reactor, the MITR-I. On June 9, 1958, the AEC issued Facility Operating License No. R-37, which authorized operation at power levels up to 1 MW. The license was effective on its date of issuance and it was to expire at midnight on May 7, 1996. Initial criticality was achieved on July 21, 1958. Supporting documentation for the above dates is contained in a letter dated June 9, 1958, from Mr. H. L. Price, Director, Division of Licensing and Regulation (AEC) to Mr. James McCormack, Vice-President (MIT).

Table 16-1

Chronology of Facility Operating License No. R-37

		Power Level	
Date	Event	Authorized	Actual
May 7, 1956	Construction Permit CPRR-5 issued.	None	None
June 9, 1958	Facility Operating License No. R-37 issued retroactive to May 7, 1956.	1 MW	None
July 21, 1958	Initial criticality of MITR-I.	1 MW	<10 kW
July 1958 - June 1959	Startup testing.	1 MW	Various
June 1, 1959	Routine operations at 1 MW initiated.	1 MW	1 MW
June 20, 1961	Amendment No. 4 issued. This authorized operation at 2 MW.	2 MW	1 MW
November 20, 1961	Operating power increased to 2 MW.	2 MW	2 MW
October 12, 1965	Amendment No. 7 issued. This authorized operation at 5 MW.	5 MW	2 MW
November 1, 1965	Operating power increased to 5 MW.	5 MW	5 MW
April 9, 1973	Construction Permit CPRR-118 issued for modification of the MITR to a light-water cooled and moderated facility.	5 MW	5 MW
May 24, 1974	MITR-I shut down.	None	None
July 23, 1975	Amendment No. 10 issued. This authorized operation of the modified reactor at 5 MW.	5 MW	None
August 14, 1975	Initial criticality of the modified reactor (MITR-II).	5 MW	<10 kW
August-December 1975	Startup testing.	5 MW	Various
December 4-5, 1975	Initial operation at 1 MW and performance of thermal power calibration.	5 MW	1 MW
February - March 1976	Core reconfigured.	5 MW	None
April 15, 1976	Routine operation at 2.5 MW initiated.	5 MW	2.5 MW
November 1976	Core redesigned.	5 MW	None
December 1976	Routine operation at 5 MW initiated.	5 MW	5 MW
February 8, 1995	License expiration date extended to August 8, 1999.	5 MW	5 MW

b) The MITR-I was operated infrequently at low power from the date of initial criticality on July 21, 1958 to June 1, 1959. The purpose of this lower-power operation was to conduct startup testing pursuant to the operating license requirements.

c) The authorized operating power level was increased to 2 MW on November 20, 1961. It was increased to 5 MW on October 12, 1965.

d) On April 9, 1973, Construction Permit No. CPRR-118 was issued to MIT by the AEC. This permit authorized modification of the MITR. At 4:18 p.m. on May 24, 1974, the original MITR-I was shut down for the last time and further operation was precluded until the construction specified in Permit No. CPRR-118 was complete. The MITR-I was heavy-water cooled and moderated. It had been decided to modify this reactor so that it would be light-water cooled and moderated with a heavy-water reflector. The new design, known as the MITR-II, offered higher flux levels for the same power as well as significantly reduced tritium production. On July 23, 1975, the U.S. Nuclear Regulatory Commission (NRC) issued Amendment No. 10 to Facility Operating License No. R-37. This amendment authorized operation of the modified reactor at power levels up to 5 MW. The amendment was effective on its date of issuance and did not alter the expiration date (May 7, 1996) of the license as originally issued. Initial criticality was achieved on August 14, 1975. Supporting documentation for the above dates is contained in a letter dated July 23, 1975, from Mr. George Lear, Chief, Operating Reactors Branch #3, Division of Reactor Licensing (USNRC) to Mr. Lincoln Clark, Jr., Director, MIT Research Reactor (MIT).

e) The modified reactor, the MITR-II, was operated infrequently at low power from the initial criticality on August 14, 1975 to April 15, 1976 when routine operation at appreciable power (2.5 MW) became possible. The purpose of this low-power operation was to conduct startup testing, much of which is documented in the report, "MITR-II Startup Report." This report was prepared for the USNRC pursuant to license requirements. During the course of this testing, axial and radial power mappings were performed. Low power testing continued until March 8, 1976 and routine operation at 2.5 MW was achieved in mid-April 1976. Operation at the authorized power level of 5 MW was achieved in November 1976.

f) On February 8, 1995, the U.S. Nuclear Regulatory Commission authorized an extension in the expiration date of Operating License No. R-37 to August 8, 1999. This was done (as is standard industry practice) to allow MIT to recover time spent in construction or modification of the reactor when the reactor was unavailable for use.

The MITR-I generated 10,435.2 MWD of energy. The MITR-II produced 18,855.4 MWD as of August 8, 1999.

16.3 Component Assessment

An assessment is provided here of the effect, if any, that prior use of reactor components or systems could have on the capability of these components and systems to perform their intended safety functions. Two factors that bear on this issue are:

a) Prior Use at Other Facilities: The MITR currently does not utilize any components or systems that have been subject to prior use at other facilities.

b) Maintenance Program: All MITR components and systems are covered by a maintenance program that provides for both regular testing of operability and repair/replacement of any degraded part. See Section 16.4 of this report.

16.3.1 Mechanical Components

16.3.1.1 Containment Building

A complete description of the MITR containment building is given in Chapter 6 of this report. To summarize, the MITR is equipped with a full containment that serves as the third and final barrier against fission product release. The major penetrations are the main and basement personnel airlocks, the intake and exhaust ventilation ducts, and a number of small lines for CO_2, helium, and compressed air. In addition, there is an airlock of sufficient size to accommodate large vehicles. The containment building is protected against both under- and overpressure. The former is achieved through doubly redundant sets (i.e., two sets each with two breakers in series) of vacuum breakers. These open to admit air into the building in the event of an underpressure. Overpressure protection is provided by a pressure relief system that must be manually operated.

The principal mechanism that could cause deterioration of the containment building is corrosion of the steel shell. In particular, galvanic attack below ground could result in pitting and above ground in a more or less uniform rusting of the surface. The mechanism would, if present, cause a very slow decrease in building capability. The material condition of the containment building is maintained below ground level through the use of sacrificial zinc anodes (cathodic protection system). Protection above ground is provided by periodic painting of the building.

The MITR Technical Specification entitled, "Reactor Containment Integrity and Pressure Relief System" pertains to the containment building. It specifies the allowed building leak rate (1% of the contained volume per day per psi overpressure), the efficiency for iodine removal of the pressure relief system filters (95%), the setpoint of the interlock that precludes reactor startup unless the building is slightly below atmospheric pressure (0.1 inches of H_2O), the setpoint for the vacuum breakers (atmospheric pressure 0.1 psig above building), and the setpoint for the building overpressure scram (3.0 inches of H_2O). Other specifications specify surveillance frequencies. Table 16-2 lists surveillance tests that are performed both to ensure compliance with the technical specification and to detect any incipient degradation of containment capability. The conduct of the building pressure test, which is discussed below, illustrates this process.

Each of the containment building's penetrations such as the airlocks, ventilation ducts, and vacuum breakers contains two closure devices that are mounted in series. Accordingly, the building pressure test is run in two different configurations so as to test each closure device separately. The test is performed by first placing the building in the desired configuration and then pressurizing the building to 50 inches of H_2O (~2 psig). The volume of the air added to the building in order to maintain 50 inches of H_2O over a period of several hours is measured as are temperatures throughout the building, relative humidity, the cycling of the reactor gasholders, and the barometric pressure. From this information, both the observed building leak rate and the total leakage are calculated. These are then compared to the allowed values. In addition, comparisons are made with data from previous test results so that any trends are identified.

The operating record for the MITR containment building is excellent with the measured leakage (i.e., the test results) for the building always being well below that which is allowed.

Table 16-2

MITR Surveillance Tests Relevant to the Containment Building and/or Its Penetrations

Procedure #	Title	Frequency
6.1.2.1	Building Pressure Test	Biennial
6.1.2.2	Main Ventilation Damper Inspection	Annual
6.1.2.3	New or Repaired Containment Penetration Leak Test	As needed
6.1.2.4	Test of Vacuum Breaker Set Points	Annual
6.1.2.5	Charcoal Filter Efficiency Test	Annual
6.1.3.5	Building ΔP Indicator and Recorder Calibration	Annual
6.1.3.6	Building Overpressure Scram	Annual
6.1.4.3	Damper Closing Time	Annual

16.3.1.2 Ventilation System Dampers/Gaskets

The damper seals in the ventilation ducts and the gaskets in the various personnel locks are made of rubber. The principal deterioration mechanisms are mechanical wear and loss of ductility. The former can result from excessive testing of the damper closure mechanism. The latter is the result of age. Both the damper seals and the gaskets are inspected periodically as shown in Table 16-2. These inspections and tests ensure that any degradation is detected early. All dampers and airlock doors are redundant. Replacement seals/gaskets are purchased from the original manufacturer if possible. If not, the original specifications are used. These ensure that rubber of the proper durometer (hardness) is used. These items are utilized in accordance with the original manufacturer's directions.

16.3.1.3 Cathodic Protection

The underground portion of the containment building is protected against corrosion by the use of sacrificial zinc anodes. A low d.c. voltage difference is maintained between these buried anodes and the steel shell of the containment building so that any galvanic corrosion will occur to the anode and not the shell. Based on the recommendation of an independent consulting firm, the entire original cathodic protection system was replaced in 1994. The system is maintained and operated in accordance with the manufacturer's directions.

16.3.1.4 In-Core Components

Deterioration mechanisms include corrosion and radiation damage. The former is discussed here. The latter is addressed in Sections 16.3.1.5 and 16.3.1.6 of this report, respectively. Many types of corrosion exist. The most prevalent is general corrosion which is defined as a uniform, slow degradation of a surface. Another type that might conceivably occur in research reactor environments is chloride stress corrosion, which requires high chloride ion concentration, oxygenated water, and high temperature. Protection against corrosion is achieved by proper control of water chemistry, particularly pH and conductivity. For aluminum-based systems (primary and D_2O), a slightly acidic pH is desired. For iron-based systems (secondary) a basic pH is desired.

a) Fuel - The fuel is designed in accordance with the specifications described in Sections 3.1.1.1 and 3.5(a) of this report. Among those criteria is one on fission density. Observation of this limit provides protection against excessive oxide layer buildup and fuel swelling.

b) Cladding - The cladding is the first barrier against fission product release. Failure could occur as the result of the above-mentioned corrosion mechanisms, an excessive fission density that causes swelling because of the build up of fission product gasses, or a manufacturing defect. Operating experience with MITR fuel elements has been excellent. A total of 153 elements (43 Gulf United Nuclear Fuels, 40 Atomics International of Rockwell International, 70 Babcock & Wilcox (as of 4/99)) have been cycled through the MITR core. There have been several incipient clad failures as documented in reports to the U.S. Nuclear Regulatory Commission. However, none were attributable to corrosion or swelling. It should be noted that a clad failure would not result in a catastrophic release of fission products to the primary coolant because the fuel is a cermet with a specified porosity. Fission products would diffuse out.

c) Metal Surfaces - The core housing, light-water tank, heavy-water tank, and all in-core components such as the natural circulation and anti-siphon valves were newly installed during the reactor modification in 1974. Thus, even though Facility Operating License No. R-37 dates to 1958, the core components themselves were new in 1974/1975. As discussed in Section 16.4 of this report, these components are inspected quarterly. No deterioration has been noted.

d) Electro-Magnets - Electro-magnets are used to connect the reactor's shim blades to the shim blade drives. These magnets, which are physically immersed in the reactor coolant, are electroplated to minimize corrosion. Nevertheless, the surfaces do eventually corrode. The present set of electromagnets was installed in 1988. One was replaced in 1994 because of surface corrosion. The others remain in excellent condition. Their condition is checked quarterly by both visual inspection and by measuring the resistivity of the coils. A failure would result in a passive reactor shutdown because the magnet must be energized to hold up its associated shim blade.

16.3.1.5 Neutron Absorbers

The absorbers for the shim blades are made of boron-impregnated stainless steel. The absorber for the regulating rod is made of cadmium. Deterioration of the shim blades is a concern because of the $^{10}B(n,\alpha)^{7}Li$ reaction which produces helium gas and hence could cause swelling of the blades. The potential consequence would be that a blade becomes inoperable. Protection is provided against this scenario by the use of six blades and by the provision for reactor shutdown by dumping the heavy water reflector.

For much of the life of the MITR-II (1976-1997), the shim blades were checked for swelling by running a fit gauge over the blade. This was done at six month intervals. No swelling or other dimensional change was ever detected. This testing was discontinued in 1997 as part of an ALARA review.

Another potential issue associated with neutron absorbers is burnup. Measurements of the reactivity worth of the MITR's absorbers are done annually. However, in order to assess the potential effect of burnup more accurately, a cross-calibration was done of an in-core absorber and a new absorber for the first few years of operation of the MITR. (The new absorber would be installed for the test and removed upon its completion.) This showed that the loss of integral reactivity worth was almost linear with energy production [16-1]. Given this result, MITR absorber elements are replaced about every 125,000 MWH.

16.3.1.6 Light-Water Core Tank

Aluminum does not undergo a ductile-brittle transition as do ferritic steels. However, damage can occur because of irradiation aging (in which aluminum transmutes to silicon), corrosion, and low-cycle fatigue. These three damage mechanisms are pH, the stress to which the core tank is subject, and the fast and thermal neutron fluences. The former may be discounted entirely because the pH has been kept slightly acidic so as to minimize corrosion. The total stress level (thin wall formula) on the core tank is approximately 60 MPa. The endurance limit (i.e., the stress level that the material can withstand for an indefinite number of cycles without failure) of aluminum 6061 alloy is approximately 200 MPa [16-2]. There is, therefore, a margin of at least a factor of three. The effect of fluence on aluminum may also be discounted. There are three reasons for this conclusion. First, the MITR core housing, core tank, and reflector tank were newly installed in 1974/1975. Second, both the fluence and the fast-to-thermal flux ratio for the

MITR's aluminum vessels are bounded by those of vessels at other facilities such as the High Flux Beam Reactor at the Brookhaven National Laboratory. Third, the effect of fluence on aluminum has been studied by Weeks, et al., who derived empirical correlations that give tensile strength and elongation as a function of fluence [16-3]. Information is also given by Farrell and Richt [16-4]. These can be used to estimate the changes that will occur in these properties for the MITR light-water core tank. This was done by Huang who showed that the material properties of the light-water core tank would be satisfactory if the MITR were to be operated for an additional twenty years at 10 MW [16-2].

16.3.1.7 Exterior Surfaces

The reactor core tank is surrounded by the heavy water (D_2O) reflector tank which is in turn encased by the graphite reflector. Hence, it is not possible to inspect or observe the D_2O reflector tank on a routine basis. The inner surface of the D_2O tank is, of course, exposed to heavy water. The quality (pH, chloride, conductivity) of that water is monitored and a deuterated mixed-bed ion column is used to maintain its purity. Hence, no corrosion would be expected on the inner surface of the D_2O tank. The outer surface adjoins the graphite reflector which is kept under a blanket of inert (helium) or non-reactive (CO_2) gas. Hence, no corrosion would be expected on the outer surface of that tank.

Three inspections have been conducted of the graphite reflector and the exterior surface of the D_2O tank. One was performed in 1986 and the others in 1987 and 1989. In all cases, access was achieved by first removing one of the experiment ports that extend downward into the graphite region and then installing a periscope in the vacated space. In addition, scrapings were obtained from the surface of the tank.

The first inspection was performed over the period of December 1, 1986 through December 29, 1986. At that time, the exterior surface of the reflector tank was found to be uniformly covered with a corrosion layer. This layer appeared to be only a few mils deep. Samples showed it to be a white powder that had a granular consistency. It was possible to make a scratch in this layer by using an aluminum tool. A γ - spectrum and other analysis of samples showed it to be aluminum oxide with trace impurities of chromium, cobalt, iron, zinc, and potassium. A small area of the reflector tank surface was then polished so as to remove all traces of the oxide layer.

The second inspection of the outer surface of the reflector tank occurred about a year later, late in 1987. The original oxide layer was still present. The area of the tank that had been polished was examined and no visible changes were noted. In particular, there was neither any visible change to the polished area nor any oxide buildup on it.

The results of both inspections were provided to the MIT Reactor Safeguards Committee (MITRSC) and to several members of the MIT faculty who specialize in corrosion. The conclusion of the MITRSC was as follows:

> "It is believed that this oxide layer formed shortly after the initial operation of the MITR-II in 1975. A small amount of moisture may have been present in the graphite reflector region and it would have condensed on the outer wall of the reflector tank. In any event, the layer is stable and not growing. Also, it is only a few mils thick." [16-5]

A third inspection of the graphite region was made in December 1989. The findings were identical to those in 1987. No further inspections have been made. It should be noted that this oxide layer may have a beneficial effect in that it may passivate the metal surface.

16.3.2 Electronic Components

Electronic systems are less subject to deterioration than mechanical ones because most are located in environmentally controlled areas. The principal concern is the breakdown of insulation. Detection is provided by test and calibration checks as well as regular maintenance.

MITR operating experience is that many electronic systems become obsolete before the equipment exceeds its useful life. For example, spare parts (especially analog ones) may cease to be available. This is a problem common to many industries. Several MITR electronic systems have recently been replaced for this reason. These include the radiation area monitoring and the radiation effluent monitoring systems.

16.3.2.1 Nuclear Instruments and Cables

The neutron/gamma radiation detectors and associated signal cables are located near the core and hence subject to radiation damage. Failure of a given detector is not an immediate safety concern because there are multiple safety channels as described in Chapter 7 of this report. The condition of the detectors and cables is checked through a variety of methods. These include:

a) Low voltage circuits continuously monitor the voltage applied to the detectors used in the nuclear safety system.

b) Operability checks are performed of all detectors at least monthly.

c) Plateau curves are determined annually for all detectors.

d) Detector cables for the nuclear safety system are checked annually for resistance (i.e., insulation satisfactory).

16.4 Preventive and Corrective Maintenance Program

Mechanical and electronic maintenance is the responsibility of the Heads of the Reactor Maintenance and Reactor Instrumentation sections, respectively. These individuals report to the Superintendent of Operations and Maintenance as shown in Figure 12-1.

Both regular preventive and corrective maintenance are practiced.

16.4.1 Regular Preventive Maintenance

If a component is important to reliable operation and/or safety and if it is subject to possible wear, then it is scheduled for preventive maintenance on a regular basis. One example is the shim blade and regulating rod drives. These, together with the shim blade magnets, are removed, examined, and rebuilt at the same time that the blade and rod absorbers are replaced. (See Section 16.3.1.5 of this report.) A second example is the gaskets for the building ventilation dampers. These are inspected every six months. If deterioration is found, they are replaced. A third example is the pressure relief system's charcoal filters. Air is drawn through these filters monthly in order to maintain their capacity to absorb iodine.

The basis for selecting the interval at which preventive maintenance is performed depends on the component. For items purchased from a vendor, the manufacturer's recommendations are followed. For items designed at MIT, past operating experience is used as guidance.

Three methods are used to schedule preventive maintenance. The first method is to monitor the energy production of the core. This applies to components such as the blade drives which are rebuilt whenever an absorber is replaced. Their replacement in turn depends on the MWH produced since their installation. The second method is to observe a master schedule. This applies to components such as the ventilation damper gaskets. These are inspected at a preset frequency. The third method to incorporate the maintenance in a checklist that is done at a frequency equal to or greater than that needed for the maintenance. This applies to the pressure relief system filters.

16.4.2 Corrective Maintenance

Corrective maintenance is practiced through use of a "job workbook." Any malfunction or observed deterioration in performance of an item of equipment is entered in this workbook and assigned a job number. The supervisor in charge then reviews the situation, conducts diagnostic tests, orders repair parts as appropriate, and schedules the repair. Once the repairs are completed, testing is required to be performed satisfactorily prior to the final sign-off of the work. The entire process is audited quarterly by the Quality Assurance Supervisor or designate.

References

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16-4 K. Farrell and A.E. Richt, "Postirradiation Properties of the 6061-T6 Aluminum High Flux Isotope Reactor Hydraulic Tube," ASTM STP 570, 1975.

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Chapter 17

Other License Considerations: Medical Use

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Chapter 17

Other License Considerations: Medical Use

17.1 Medical Use of the MITR

The authors of the Atomic Energy Act of 1954, as amended (the AEA), recognized the potential uses of special nuclear material in medical therapy. Accordingly, Section 104a of the AEA authorizes licenses to be issued for the utilization of research reactors in medical therapy. These potential uses were also recognized by the designers of the original MIT Research Reactor and MIT therefore obtained licenses for the facility under both Sections 104a and 104c of the AEA.

There have been many projects performed at the MITR since its initial criticality in 1958 that have had a medical benefit as their objective. Examples include the development of osmium-iridium generators for use in diagnostics, the study of mineral uptake in the human body using the stable isotope method, the investigation of possible links between chemical imbalance in the brain and heart attacks, and radiation synovectomy which is the use of beta-emitting isotopes to treat arthritis of the knee. Most of these projects have involved either the production of isotopes for subsequent use at a hospital or the analysis of samples such as blood or urine that have been obtained from subjects at a hospital. These applications of the MITR are licensed under Section 104c of the AEA. Section 104a refers to the direct use of the MITR for medical therapy. Thus far only one application, Boron Neutron Capture Therapy (BNCT), has been conducted under this section.

BNCT involves the pre-irradiation administration of a boron compound to the patient. The compound is chosen so that it will selectively concentrate in diseased tissue. Once this has happened, the diseased area is exposed to a beam of neutrons. This causes the boron to fission. The resulting lithium and helium nuclei are heavy charged particles that have substantial kinetic energy. Their range is short and thus the energy that is locally imparted is high. The result is that diseased cells are killed and adjacent healthy cells are spared. An essential factor in the conduct of

BNCT is, therefore, for the boron to become concentrated in the diseased tissue, preferably in the cell nuclei.

BNCT was first attempted with thermal neutron beams at the Brookhaven National Laboratory's Graphite Reactor in the late 1950s and at the MITR-I in the early 1960s for the treatment of glioblastoma multiforme, a highly malignant form of brain tumor. Untreated victims of the disease have a life expectancy of 4-5 months. With conventional therapy (steroids and X-rays), life expectancy is a year or less. Much was learned from these initial trials and later the clinical research effort was continued in Japan. The information gained from the original trials and the continuing Japanese effort, together with the technical advances that were made in the ensuing decades, enabled a new series of trials to be initiated at both BNL and MIT in the mid-1990s and more recently at Petten and Finland in Europe. Major differences between the present effort and the original one include the following:

a) An epithermal neutron beam is used instead of a thermal one. This provides for penetration of the neutrons through the intact skull to the mid-line of the brain. Thus, deep-seated tumors can be targeted without the necessity of surgery to remove the skull and scalp and thereby expose the tumor.

b) Reactor physics methods improved as did computational capabilities. Thus, improved estimates of delivered dose are now possible.

c) Superior boron-containing compounds were identified for improved concentration in cancerous tissue.

d) Methods to identify the location of boron atoms in cells were developed.

e) Another type of cancer, metastasized melanoma, has also been targeted for treatment using BNCT.

Theoretical work on BNCT has occurred more or less continuously since the idea was first proposed by Gordon Locher in 1932. However, there were no patient trials in the U.S. between the end of the original BNL/MIT effort in the early 1960s and 1994. In contrast, the use of other forms of radiation therapy, particularly teletherapy and brachytherapy, became routine. As a result, regulation of BNCT (and other possible Section 104a applications) did not keep pace with that for other techniques. Thus, in the early 1990s when MIT decided to initiate a Phase-I trial of

BNCT using an epithermal neutron beam, it was necessary to formulate the appropriate regulations. This necessity resulted in the issuance of MITR Technical Specification No. 6.5, "Generation of Medical Therapy Facility Beam for Human Therapy." This effort was guided by 10 CFR 35 (Subpart I, Teletherapy). However, the use of neutron beams differs in many ways from that of teletherapy.

MITR Technical Specification 6.5 was written conservatively, particularly in terms of calibration frequency requirements. It is expected that these and other sections of the specification will be reviewed as experience is gained.

17.2 Responsibilities for Use of Medical Therapy Facility Beams

The use of a medical therapy beam that is generated at a research reactor for patient treatment requires the coordinated efforts of both a medical organization such as a research hospital and an engineering- and science-oriented organization such as a research university. Both organizations must be licensed by the U.S. Nuclear Regulatory Commission or the corresponding agency for an Agreement State and both organizations have certain responsibilities to ensure the proper application of the beam. The division of responsibility for use of the MITR's beams in patient therapy is as follows:

a) All medical treatments, including irradiations, treatment planning, and analysis of the neutron capture agents in patients, are the responsibility of the BNCT physician authorized user in charge of the therapy and the medical physicists from the NRC-licensed or Agreement-State licensed medical center.

b) MIT is responsible only for providing current and accurate beam characteristic parameters to the medical use licensee and for the delivery of the desired radiation fluence as requested in the written directive.

Under this division of responsibility, the medical-use licensee (the hospital) is responsible for the patient, preparation of a treatment plan and written directive (as defined in 10 CFR 35), administration of the boron pharmaceuticals and the neutron beam to the patient, supervision of the setup and irradiation of the patient, control of the byproduct materials found in the patient's body as a result of the treatment, and adherence to a quality management program for the conduct of human therapy. MIT is responsible for providing the medical therapy treatment facility, production of the neutron beam, physical characterization of the beam at its interface with the patient, all health physics considerations associated with the beam (except for the patient), control of radioactive contamination and activation of the medical therapy treatment room and its contents, and adherence to a quality management program for the conduct of human therapy.

It should be noted that either the medical use licensee or MIT may have other regulatory obligations pursuant to requirements of the U.S. Food and Drug Administration. However, these are beyond the scope of this document. Also of significance is that the U.S. NRC does not regulate the design of the beam. The characteristics of a beam necessary for treatment are decisions that are medical in nature.

17.2.1 Internal Oversight

Both the medical use licensee and MIT have internal committees with oversight responsibility for the use of medical therapy facility beams for patient therapy. For MIT, these organizations are the MIT Committee on Reactor Safeguards, the Committee on the Use of Humans as Experimental Subjects and the Committee on Radiation Exposure to Human Subjects. As shown in Figure 12-1, the Chairmen of these Committees report directly to the President of MIT. For the medical use licensee, there are Institutional Review Boards (IRBs) that fulfill the same functions.

17.3 Regulatory Commitment

17.3.1 Basis of Commitment

MIT's commitments concerning the generation of medical therapy beams for patient therapy are contained in the cognizant technical specification. The reasoning behind each provision of the original (1993) specification is summarized here. Reference should be made to the actual specification when reading this material. Also, there have been some changes to the specification since 1993. For example, provision (19), which concerns calibration of reactor facilities that are used to measure boron concentration in blood, has been added.

As noted earlier, the use of beams from research reactors to treat patients is not addressed in any detail in the Code of Federal Regulations. The most relevant material is to be found in Subpart I, Teletherapy, of 10 CFR 35, "Medical Use of Byproduct Material." Those requirements were used in part to develop the MITR technical specification. However, a one-to-one correspondence was neither appropriate nor possible. Subpart I concerns the use of Co-60 and Cs-137 sources and not the use of a research reactor to produce a beam of radiation. There are many differences between an operating reactor and an installed gamma-ray source. These include:

a) The reactor can be shut down or scrammed thereby greatly reducing the intensity of the beam. This feature makes use of a research reactor to produce a beam far less hazardous to the patient than the use of a sealed gamma source. Radiation from the latter can be halted only by the use of shields. In contrast, the intensity of the reactor's beam can be greatly reduced either with shutters (shields) or by a reactor shutdown action.

b) The reactor is heavily regulated and has NRC-approved quality assurance, training, and radiation protection programs in place.

c) The Reactor Staff is committed under the provisions of the reactor's operating license to report any instance of non-compliance with its technical specifications. In contrast, medical licensees are required to report only those items that are listed in 10 CFR 35 as being reportable.

Each provision of Technical Specification 6.5 (as it was issued in 1993) is listed below together with a reference to the relevant requirement of either Subpart I of 10 CFR 35, 10 CFR 35.33, or 10 CFR 35.2. Whenever a match is not appropriate, the reason for it is listed as reactor-specific.

10 CFR 35	Technical Specification 6.5
Reactor Specific	Patients are accepted only in accordance with a written directive from an NRC or an Agreement State Medical Use Licensee which is licensed for use of the MIT Research Reactor's Medical Therapy Facility beam for human therapy. (Provision (1))
Reactor Specific	Provision (2) delineates division of responsibility between MIT and the referring medical licensee. Also, it establishes both the authority and protocol for initiating and terminating a radiation therapy.

Reactor Specific	Provision to scram the reactor thereby greatly reducing the beam as a source of radiation. (Provision (3))
35.615(a) Access Control	Provision (4) of the specification.
35.615(b) Interlocks	Provision (5) of the specification. Subprovisions (a) and (b) of the specification correspond to subclauses (b(1)) and (b(2)) of 10 CFR 35.615. Subclause (b(3)) is not applicable because there is no "reset" for a reactor beam. Subprovisions (c) and (d) of the technical specification impose design requirements on the shutters that are more conservative than Part 35.
35.615(c) Indicator Lights	Provision (6) of the specification.
35.615(d) Radiation Monitor	Provision (7) of the specification. Subclauses (1), (2), (3), and (5) of the 10 CFR 35.615(d) are directly addressed in the subprovisions of the specification. Subclauses (4) and (6) are not so addressed because these items are covered as part of the MIT Research Reactor' radiation protection and maintenance programs.
Reactor Specific	Provision (8) of the specification establishes a means for two-way communication between personnel at the medical therapy facility's local control panel and the reactor operator stationed at the reactor control room console.
Reactor Specific	Provision (9) of the specification imposes a requirement for the manual opening of the door to the medical therapy facility. This requirement is necessary because the door also acts as a shield and is designed so as to be movable without electric power in the event of a power failure.
35.615(e) Patient Observation	Provision (10) of the specification.
35.2 Clauses 4(iii) and 4(iv) of Definition of Misadministration	Provision (11) of the specification establishes criteria for the accuracy of the radiation fluence delivered by the medical therapy facility beam. This provision represents another difference between the use of a reactor beam and a sealed source for medical therapy. The reactor treatment is a two-step process involving delivery of a fluence and a subsequent nuclear reaction in the tumor based on the boron concentration in tissue. It is this nuclear reaction that generates a major portion of the dose. In contrast, the use of a sealed source is a one-step process with the dose in the tumor being that delivered by the source. Provision (11) was inserted in the specification to establish allowable limits on the delivered radiation fluence. Issues of misadministration and recordable events as they concern generation of the medical therapy facility beam are delineated in definitions (7) and (8) of this specification.
Reactor Specific	Provision (12) of the specification establishes criteria for the testing of all interlocks and channels. (Note: Testing of the radiation monitor is addressed separately in provision (7(b)).

Reactor Specific	Provision (13) of the specification establishes criteria for testing the manual operation of the facility's shield door.
35.632/35.634	Provision (14) of the specification specifies frequencies for full characterization and calibration checks of the beam. These terms, which are defined in the specification, correspond roughly to the words "full calibration measurements" and "Periodic spot-checks" in 10 CFR 35.632 and 35.634 respectively. However, there are major differences. A sealed source provides one type of radiation. The reactor beam has fast neutron, thermal neutron, and gamma components. While the reactor beam has many components, its characteristics correlate with the reactor's power level making it relatively easy to reproduce from day to day. It should be recognized that the criteria listed in 10 CFR 35.632 and 10 CFR 35.634 are specific to sealed sources and cannot be applied directly to a reactor-produced beam. (Note: This provision was revised in 1999 to define functional checks, calibration checks, beam characterizations, instrument calibrations traceable to NIST, and the minimun number of beam monitors.)
35.605	Provision (15) of the specification provides for the proper supervision of all maintenance and repair activities. As such, it fulfills the same function as 10 CFR 35.605.
35.610	Provision (16) of the specification provides for the training of non-licensed personnel to operate the medical therapy facility beam. Also, it provides for the posting of instructions to be followed in the event of an abnormality. This provision is similar in many respects to 10 CFR 35.610.
35.33	Provision (17) of the specification provides for the recording of "recordable events" and the reporting of "misadministrations."
35.2 Misadministration	Definition (9).
35.2 Recordable Event	Definition (8).
35.2 Written Directive	Definition (10).

17.3.2 Content of Commitment

The content of MIT's commitment to the safe generation and use of medical therapy beams for human therapy is enumerated here against the guidelines of NUREG 1537, Rev. 0, February 1996.

17.3.2.1 General Criteria

General criteria for the performance of medical therapy using neutron beams are enumerated here:

a) Provision (1) is a commitment to deliver neutrons to treat patients only pursuant to a written directive from a BNCT physician authorized user who is specifically authorized to perform BNCT by an NRC- or Agreement-State issued medical use license.

b) Provision (17) together with Definitions (8) and (9) are a commitment to record events equivalent to "recordable events" in 10 CFR 35.2, report events equivalent to "misadministrations" in 10 CFR 35.2, and establish a written quality management program for use of the neutron beam using criteria similar to those specified in 10 CFR Part 35 for teletherapy.

c) The Quality Management Program and Provision (14) which concerns beam calibration and characterization as well as instrument checks, provide the methodology to ensure that the neutron flux, fluence, and spectrum delivered to the patient are delivered as requested by the physician authorized user.

d) Provision (15) ensures that design aspects of the neutron beam delivery system that are important to patient or user safety will not be changed without proper review.

e) Provision (8) provides for two-way communication between the reactor console operator and the BNCT physician authorized user. Provision (3) provides a method for terminating the treatment exposure.

f) Provision (14) provides for spot-checks prior to beam use if an activity may have altered the beam characteristics.

g) Provision (5) prevents personnel from being accidentally exposed to the beam.

17.3.2.2 Design Features

Design features for the performance of medical therapy using neutron beams are enumerated here:

a) Provision (3) provides for the possibility of scramming the reactor from the medical therapy treatment facility control area.

b) Provision (8) provides that personnel should be able to communicate from the medical therapy treatment facility control area to the reactor control room and that they should also be able to communicate from the medical therapy treatment control area to the inside of the medical therapy treatment room.

c) The shielding design criterion is 1.25 mrem/h at 6 MW on the outer surface of the walls that form the medical irradiation rooms. This value is lower than the criterion for unrestricted access to the area exterior to the rooms. Radiation levels on the surface of the viewing windows and/or the shielded doors may be higher for certain beam qualities (e.g., a thermal beam). If so, auxiliary shielding and/or administrative controls are used to maintain exposures in accordance with the ALARA principle.

d) The interiors of the medical therapy treatment rooms are designed to minimize induced radioactivities of walls, floors, and equipment. For example, boron-containing paint has been specially prepared and applied on walls. Also, layers of boral or other shield material are installed as appropriate, on floors.

e) Provisions (5) and (6) provide for shutters and appropriate interlocks to cut the beam off. Provision (3) (reactor scram) is a backup.

f) Provision (4) provides for a shielded door.

g) Provision (10) provides for patient viewing by both a viewing port (lead glass window) and a closed-circuit TV system. This provision also requires adequate emergency lighting.

h) Provision (7) provides a radiation monitoring system with visual and audible alarms both at the therapy control panel and in the therapy room itself. A capability to bypass the these alarms during treatment is provided. This provision also requires a backup power supply.

i) Provision (11) provides for the use of on-line beam monitors that determine the neutron fluence being administered to the patient. Epithermal neutrons, thermal neutrons, and gamma rays are monitored.

17.3.2.3 Administrative Requirements

Administrative requirements for the performance of medical therapy using neutron beams are enumerated here.

a) Provision (1) requires that patients be referred for treatment by a written directive from a medical use licensee authorized by NRC or an Agreement State to use the non-power reactor to provide the irradiation.

b) Provision (2) delineates the responsibility of the non-power reactor licensee and of the BNCT physician authorized user as described in Section 17.2 of this report.

c) Provision (2) states that both the non-power reactor licensee and the medical use licensee shall agree that a patient irradiation can be started. It also states that either the medical use licensee or the non-power reactor licensee has the authority to terminate a patient irradiation.

d) Written internal procedures, which have previously been provided to the U.S. NRC, provide for the removal of the patient from the medical therapy treatment room in the case of an emergency. These procedures are practiced periodically by the MITR staff.

e) Provision (3) provides for the possibility of scramming the reactor from the medical therapy treatment facility control area.

f) Provision (6) provides for the use of alternate methods of indication of shutter function. The one used would probably be an observed correlation between shutter position and radiation levels in the medical therapy treatment room. Therapy is always initiated and terminated by opening and closing the shutters in a specific sequence. Thus, the expected change in general area radiation level is known. If this change were not observed, it could be concluded that a shutter malfunction had occurred.

g) Provision (7) provides for a radiation monitor in each medical therapy treatment room. (Note: These monitors are NOT part of the area radiation monitoring system described in Section 7.7.1 of this report.) The provision provides for the calibration of this monitor, operability checks prior to use, and the use of a backup (portable detector) if needed. Alarm setpoints are also specified.

h) Provision (10) specifies that if one of the redundant methods of viewing the patient fails during an irradiation, the irradiation may continue at the discretion of the BNCT physician authorized user.

i) Provision (11) specifies limits on allowed radiation fluence relative to that specified in the patient treatment plan. Definitions (8) and (9) define recordable events and misadministrations. These definitions also specify reporting requirements.

j) Provisions (12) and (13) specify periodic surveillance requirements for the scram capability, beam cutoff interlocks, shutter status lights, radiation monitor, and intercom.

k) Provision (14) and Definitions (2) through (4) specify calibration checks, functional checks, and characterizations.

l) Provision (19) specifies that the prompt gamma facility, as well as any other reactor facility that is used to determine the concentration of boron in blood, be calibrated annually and that a one-point check be performed prior to each patient irradiation.

m) Provision (15) specifies that a senior reactor operator shall supervise repairs, maintenance, and modifications to the medical therapy facilities. This provision also defines medical equipment that is excluded from this requirement.

n) Provision (16) discusses the training that is required of personnel who are not licensed MITR reactor operators and who wish to operate the facility.

o) Provision (18) is a requirement to adhere to the Quality Management Program.

Written procedures exist to implement the provisions described above and also in Sections 17.3.2.1 and 17.3.2.2 of this report. These procedures were provided to the U.S. Nuclear Regulatory Commission as an item of information prior to the start of patient trials in 1994.