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MITR-III: Upgrade and Relicensing Studies for the MIT Research Reactor

Report No. MITNRL-061

Second annual report of redesign
and relicensing activities related to
the MIT Research Reactor

August, 1994

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Preface

The current operating license of the MIT research reactor will expire on May 7, 1996 or possibly a few years later if the US Nuclear Regulatory Commission agrees that the license period can start with the date of initial reactor operation. Driven by the imminent expiration of the operating license, a team of nuclear engineering staff and students have begun a study of the future options for the MIT Research Reactor. These options have included the range from a major rebuilding of the reactor to its decommissioning. This document reports the results of a two year intensive activity which has been supported by a \$148,000 grant from the USDOE contract Number DEFG0293ER75859, approximately \$100,000 of internal MIT funds and Nuclear Engineering Department graduate student fellowships as well as assistance from international visiting scientists and engineers.

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1 EXECUTIVE SUMMARY

1.1 Background of MITR Project

The MITR project was initiated in the early 1950's through the vision of then MIT President James Killian. He brought Manson Benedict to MIT to start a Department of Nuclear Engineering. Professor Benedict in turn brought Theos J. Thompson to MIT to design, construct, and direct the MIT Research Reactor. The reactor was completed and put into operation in 1958 after only 3.5 years of design and construction. To assure that the MITR remained a viable and competitive facility, it was planned from its inception to upgrade it at appropriate milestones. A major upgrade involving an entirely new compact core design was accomplished in 1973-1976 under the direction of Professor David D. Lanning using MIT faculty, staff, and students. The upgrade was funded by MIT at a cost of \$3 M.

At the same time that the MITR-II was completed in 1976, the reactor was placed within the framework of an interdepartmental laboratory, and Professor Otto K. Harling was invited to MIT to direct the new Nuclear Reactor Laboratory (NRL). A major reason for the new interdepartmental laboratory framework was the need for increased utilization of the facility. Since 1976 the research volume supported by the MITR-II has increased by approximately an order of magnitude in nominal dollars. At the same time it has not been able to eliminate an operating deficit. Operating deficits plague almost all of the university research reactors (URRs) in the USA. Base federal support to alleviate deficits has never been available for URRs despite the federal tradition of such support for comparable sized facilities such as accelerators.

From the beginning of its operating history the MITR has been in the forefront as a research and educational facility. Over its 35 year life it has provided MIT with a powerful and versatile tool for research and teaching in the nuclear sciences and technologies. Recently compiled statistics show that the MITR has been used by MIT students for thesis research, totaling 193 Doctoral, 274 Masters and Engineering, and 91 B.S. theses. A total of more than 820 journal articles and 240 major technical reports covering a wide range of disciplines have been published with research carried on at MITR-I and MITR-II. A good indication of the quality of the MITR based research can be found in the 1985 USDOE survey of URR productivity. This survey included 90 major achievements or technical/scientific highlights resulting from research at MITR. This important category of noteworthy achievements or highlights is more than the sum of such achievements from all of the other approximately three dozen operating URRs in the USA.

The MIT reactor project will reach a major milestone on May 7, 1996, or at the latest 18-36 months later if a license extension request is approved, when the current license for reactor operation expires. In order to develop the necessary information for an informed decision on the future of the MIT Reactor, a study was initiated by a group of MIT faculty, staff, and students and were carried out over a period of about two years.

This report provides the results from the first two years of this study. The report provides results for several scenarios including the range of options from (1) relicensing the current reactor with the minimum upgrades needed to obtain a license and a reliable reactor system, (2) several modifications to the reactor which would be upgrades and (3) the option of reactor decommissioning.

Because the time and the resources for this study were very limited (approximately \$150,000 from the USDOE and \$100,000 from MIT) these results must be considered preliminary. A number of options or scenarios were investigated in this study and are summarized in this report. Initially, we did not wish to constrain the options which could be considered. After completion of the preliminary studies, more detailed studies have been made on the recommended option; namely to upgrade the present reactor by doubling the operating power to 10 MW.

1.2 Need for Relicensing and Upgrading

The operating license for the MIT Research Reactor expires at midnight May 7, 1996 and a complete application must be submitted to the NRC at least 30 days prior to this date. On March 31, 1994, a request was submitted to the U.S. Nuclear Regulatory Commission to extend the MITR license to April 24, 2001 for the purpose of recovering time spent in construction, modifications, and low-power testing. The NRC has not yet acted on this request and one cannot predict the outcome. However, there is precedent among the power reactor community to extend licenses to recover time spent on construction. Regardless of the outcome, the long-term continued operation requires that the reactor either be relicensed at its current operating power of 5 MW or that a new license be issued for some higher power level. The objectives of the relicensing process, as specified by the US Nuclear Regulatory Commission, are to verify the condition of the facility, to ascertain that the facility is in compliance with the current regulatory practices, and to ensure that all supporting documents are current. Relicensing also brings with it certain opportunities. First, it provides a convenient means for simplifying facility operation through the incorporation of past operating experience. Second, it is possible to upgrade the facility, e.g., to some higher power level. This opportunity is particularly attractive because preliminary studies indicate that an upgrade to 10 MW is possible for the MITR-II without changes to the reactor core such a power increase would double the available neutron flux, and greatly assist in maintaining the reactor as a world-class research facility for the next two decades.

1.3 Assessment of System Conditions

An assessment of reactor components and systems was made to determine the current status of these components and to determine whether replacements will be needed for continued or upgraded operation beyond 1996. It was determined that several components including the cooling towers, underground secondary pipes, and all cooling system pumps will require replacement. In addition, a number of critical components such as the reactor core tank and reflector tank are expected to be satisfactory and difficult to

access portions may be inspected later to determine if any degradation has occurred. A number of recommendations for component redesigns were also made.

1.4 Future Importance of MITR

Research-reactor based research will continue to be an important part of the national program in the nuclear sciences and technologies. The MIT Research Reactor is one of the largest and most capable URRs; its productivity in research and in the education of leading nuclear scientists and engineers is unmatched by any other US research reactor facility. Furthermore, the MITR is unique in that it is the only large URR at a large and prominent research university.

1.5 User Requirements

A survey of future user requirements indicates that the likely major uses of a relicensed MITR will be the same as currently and in the past:

- Neutron beam tube research (e.g., neutron scattering)
- Medical applications
- In-core irradiations (e.g., water chemistry loops and materials)
- Neutron activation analysis
- Radioactive isotope production
- Irradiations for industry (e.g., silicon transmutation doping)

1.6 Project Goals

- Obtain international information on modern research reactor facilities and users, as relevant to a future MITR-III design.
- Initiate the investigation of as many options for the MITR-III as time and resources will afford.
- Make a preliminary design review and cost analysis.
- Select a recommended option.
- Investigate the recommended option in more detail.
- Prepare a two-year project report.

Note: As originally envisioned, the project to investigate the options for the future of the MIT Research Reactor (e.g., the MITR-III Design Project) was expected to take two years. Partial funding was approved by MIT in mid-1992 and some funding for the project was obtained from DOE in the spring of 1993 and 1994. Although additional help

was received from the MIT Nuclear Engineering Department in the form of fellowships during the fall and spring of 1992-3 and 1993-94, the resources have been limited (approximately \$150,000 from DOE and \$100,000 from MIT). Nevertheless, many important results have been attained, as described in the next section (Section 1.7) and some of the detailed studies are still in progress.

1.7 Results of the First Two Years of Studies

The work-time schedule shown in Figure 1.7 was developed at the initiation of the MITR-III design review project. The project started as a formal program in June of 1992. Progress has been in close accord with this schedule.

Travel to several research reactor facilities was completed by Otto Harling and David Lanning during July 1992. The information obtained was summarized to the design group in the July 17, 1992 project meeting and relevant considerations have been incorporated into the design and user requirements included in this report.

Options that have been studied include:

- The upgrading of the MITR-II to operate at 10 MW (i.e., a factor of two above the present 5 MW operation)
- The use of a 10 MW TRIGA LEU core in the present MITR-II core tank
- The use of a 10 MW SILOE LEU core (a French design) with a D₂O reflector tank redesign
- The requirements for relicensing at the present MITR-II operation of 5 MW
- The consideration for decommissioning, a review of earlier studies

The work on the design options has been primarily carried out by graduate students under the supervision of the Nuclear Engineering Department (NED) faculty and the Nuclear Reactor Laboratory (NRL) staff. The NED faculty involved included Professors Otto Harling, David Lanning, Michael Driscoll, John Meyer, and Allan Henry. The NRL staff included Dr. John Bernard, Eddie Lau, Dr. Kwan Kwok, and Tom Newton.

The doctoral graduate students and their research projects during the first year were:

Santiago Parra:	Upgrading the MITR-II to 10 MW Operation
Thomas DeLorey:	A Triangular Mesh Nodal Code Development

The students involved in Special Projects during the first year included:

Dan Doney:	Sensitivity studies on factors for the core operating limits
Chi Kao:	MITR-II thermal hydraulic loop study for a reference case

Figure 1.7. Work-Time Schedule for MITR-III

During the first year several students completed projects related to the courses being offered in the NED. For Course 22.39 Reactor Safety and Operations (Fall 1992) these projects included:

Jyh-Tzang Hwang:	Aging of Aluminum Components
Dan Doney:	PLTEMP Sensitivity Study
Renee DuBord:	Balance of Plant Transient Code (initial investigation)
Weng-Sheng Kuo:	TRIGA Fuel Core Design
Tim Lawrence:	Impact of the New 10CFR20
Rodrigo Rubiano:	TRIGA Fuel Investigation
Lorin Selby:	Use of MITR for Neutron Beam Tube Research
Ilker Tari:	MCNP - Nodal Averaged Cross Sections

For Course 22.33 Reactor Design (Spring 1993):

Bilal Bhutta:	
Lin-Wen Hu:	Design of a Core Utilizing TRIGA Fuel
Nicole Larrier:	
Mike McComas:	

During the first year, a self-supported group of three French students came for a three month study (March through June, 1993) to work with the MITR-III Design Group:

Martine Moreau:	Design of a Core and Reflector Tank for the MITR-III
Djamel Tadjer:	Utilizing the French SILOE Fuel
David Pialla:	

Also during the spring, these graduate student projects and theses were initiated and were continued or completed during the second year of the project:

Hernan Trosman:	Computer Simulation for Transient Analysis of MITR
Mike McGuire:	Project Cost Analysis and Detailed Upgrade Studies
Everett Redmond:	An MCNP (Monte Carlo) Model of the MITR-II

Also, graduate students and their research during the second year were:

Lin-Wen Hu:	Thermal-Hydraulics of the Mixing Pool
Jeff Hughes:	In-Core Monitoring
Stead Kiger:	Fission Plate Converter Filter Design for BNCT
Loren Simon:	ELAF Fuel Core Analysis Study
Dan Doney:	Boiling Noise Detection

The students involved in Special Projects during the second year included:

Joe Bambenek:	Radioactive Effluents at 10 MW
Kevin Terry:	Shielding Study
Mike Turek:	Secondary System Design

During the second year several students completed projects related to the courses being offered in the NED. For Course 22.39 Reactor Safety and Operations (Fall 1993) these projects included:

Mathew Arno:	Detailed Secondary System Design
Tamer Bahadir:	Flux Evaluations
Andrew Barritt:	Safety System Evaluation
Lin-Wen Hu:	MITR Simulator Calibration and Upgrade
Chi Kao:	Plant Component Aging Study
Kevin Terry:	Pump Coastdown and Time-to-Scram Experiments

For Course 22.33 Reactor Design (Spring 1994), a fission converter for an epithermal beam to be used in BNCT was designed and analyzed by the following students:

Nick Levitt and Raul Valdez:	Neutronic Methods and Analysis
Kevin Terry:	Shielding and Shutter Design
Stead Kiger:	Converter Filter Design and report coordinator
Ruxandra Golinescu:	Converter Thermal-Hydraulics
Tamara Galen and Vanda Merriman:	Converter Configuration and Fuel Handling Design

During the second year, two self-supported French students came for a three month study (March through June, 1994) to work with the MITR-III Design Group:

Jean-Baptiste Dutto:	Investigation of the use of the PARET code to estimate the effect of high reactivity insertions and to establish a
Stephane Evo:	reactivity insertion limit such that the fuel temperature does not exceed 450°C.

Also, during the second year there were two visiting Engineers who worked with the project:

Sylvain Sardy (from France):	Uncertainty Evaluations for MITR-III Applications
S. Kassim (from Malaysia):	Cooling Tower Design and Secondary System Review

In addition to the graduate student work, the members of the NRL staff completed important reports and reviews concerning the requirements for relicensing. These studies included investigation of the entire reactor facility to determine the requirements for repair and replacements needed for another 20 years of reliable operation.

In summary, the project results are as follows:

- The investigations indicate that relicensing should be possible with some needs for replacement and/or repair such as the cooling towers and attendant piping and replacement of heat exchangers and all major coolant pumps.
- The TRIGA-fuel core design studies were made to investigate the potential for enhanced inherent safety combined with a commercially available fuel. It was found that although a core design could be made that would fit within the present MITR-II core tank, the capability was very limited and a useful core design would require a slightly larger new core tank. Also, the design of the fuel was such that the available neutron flux at 10 MW would be lower than the MITR-II would achieve if operated at 10 MW. Removal of the old core tank and installation of a new one implies a costly modification.
- The SILOE-fuel core design used the French fuel designed for a "general" research reactor of higher power than 10 MW. Therefore the thermal hydraulic design of the fuel is very conservative. This makes the power density and neutron fluxes lower than the MITR-II would have at 10 MW. The advantages of the design include ample space for in-core facilities and simplification of core limits for operation. However, a new core tank is required with the attendant major costs.
- Since it is expected that any new core design for the MITR-III (such as TRIGA or SILOE fuels) would be required by the NRC to be of "Low Enrichment" (i.e., less than 20% U^{235}), these design options have been studied for LEU (Low Enriched Uranium) Fuel. In general these design options are not predicted to give a flux-to-power ratio that is as high as the present MITR-II with its compact core using highly enriched uranium.
- Preliminary investigations to date all indicate that the MITR-II can be upgraded from the present 5 MW operation to run safely at about twice the power (10 MW), and the system modifications to allow this upgrade are similar to the component replacement, repair, and documentation that would be required in any case for relicensing at 5 MW.
- A more detailed engineering design analysis and review has been initiated to establish the requirements and cost of the potential relicensing with an upgrade approaching 10 MW operation. The results to date are included in this report.

1.8 Status of Cost Estimates

Preliminary cost estimates have been developed for relicensing the MITR-II at 5 MW and for relicensing and upgrading the MITR-II to 10 MW. The cost of decommissioning the MITR-II has also been estimated using as a basis a study of the decommissioning cost of MITR-II which was performed by an outside contractor, General

Electric, in December of 1988. The GE study provided a formula for estimating decommissioning costs in future years and this was used to obtain the current estimate. It should be understood that there are major uncertainties involved in the cost of decommissioning and this will be discussed.

The cost analyses included in this report were divided into three sections. The Engineering and Design Costs include the funding needed to complete the design work for a relicensed or upgraded reactor. The Operating Costs refer to the cost of running the Nuclear Reactor Laboratory (NRL) during the period of refurbishment. This includes the salaries of the technicians and other NRL employees who will be replacing and repairing reactor components. The Other Direct Costs are the costs of the new hardware. For the option of relicensing MITR-II at 5 MW several key systems will be refitted with new components of similar capacity. If a decision is made to go to 10 MW, the replacement components will be large enough to accommodate operation at the increased power level.

The Engineering and Design work for the two options (5 MW relicense or 10 MW upgrade) are similar because the MITR-II SAR will be updated in either case. Some additional research work will be needed for the upgrade case due to the higher heat removal requirements. The main reactor coolant system components (heat exchangers, pumps, cooling towers) will need to be replaced in either case. Because the cost of a component is only a weak function of its size, it was found that the Direct Costs of the 10 MW upgrade are only about 11 % higher than the 5 MW relicense. A three month downtime was assumed for the relicense, while a four month period was used in the upgrade analysis. Because the operating costs are sensitive to the estimated construction time, the operating costs are about 28 % higher for the 10 MW scenario.

Based on this report, the cost of relicensing the MITR-II at its current 5 MW power level is estimated to be \$3.22 M in 1994 dollars, while the cost of relicensing and upgrading to 10 MW is estimated to be \$4.02 M. These are best estimate totals with a contingency factor to account for unforeseen events. The contingency factor is equal to 25 % of the total best estimate cost. Modifications and costs could be distributed over an extended time period avoiding costly shutdowns. The cost of decommissioning in 1996 is estimated to be between \$14.0 M and \$20.0 M in 1996 dollars (approximately between \$12.0 M and \$17.0 M in 1993 dollars). However, the complete decommissioning and decontamination of the MITR-II may be impossible in the foreseeable future due to problems with returning the spent fuel and lack of disposal sites for the large amount of low level waste which would be generated in a complete decommissioning. See Section 3.9 for a detailed discussion of these issues.

1.9 Major Recommendation

It is recommended that the detailed engineering and safety analyses be completed such that either of the following options can be chosen:

- 1) Upgrade to 10 MW and relicensing of the MITR-II
- 2) Relicensing at the current power level of 5 MW

Emphasis is to be toward the upgrade to 10 MW, with an easy fall back position of relicensing at 5 MW.

The recommended course of action favors the maintenance of a viable research reactor at MIT. This recommendation recognizes the unique value of a good neutron source located at MIT, and its importance to the Institute and to the country's program in nuclear science and technology. In the first year of our study of options, we explored a range of possible modifications which could be used for MITR-III and which had some attractive aspects such as larger spaces for in-core facilities and the use of low enriched fuel. Finally we chose the recommended direction of a relicensing and upgrade of the existing reactor configuration because on balance it offers the best overall performance, can be implemented without any long shutdowns, and its cost is likely to be less than the cost of the TRIGA and SILOE fueled options which were studied. It is expected that the NRC will continue to permit the use of highly enriched fuel in the current reactor configuration (which is proposed for relicensing and upgrade) until a new fuel such as a high density uranium silicide fuel is developed that will allow the compact core design to be operated on "Low Enriched Fuel" (LEU). It has been the understanding and commitment to the NRC that such LEU fuel would be used at the MITR if it becomes available.

2 INTRODUCTION

The subsequent sections and appendices of this report will summarize the research that has been completed during the two years of the MITR-III study project. This research has been in the form of Master theses, Ph.D. theses, Special Problems, and MIT coursework, as well as studies conducted by members of the NRL staff, visiting engineers, and NED faculty.

Prior to the initiation of serious design work, the status of the MITR facility and its component systems were analyzed and evaluated. At the same time, a survey of reactor users was conducted to identify what capabilities will be required for their future research.

A comparison of these user requirements with the current status of the reactor led to the conclusion that the design options for the future of the MITR should be carefully studied. Due to time and resource constraints, five options were selected for in-depth analysis:

- Relicense MITR-II at its current power level of 5 MW (Section 5)
- Upgrade to 10 MW with the same core configuration (Section 4)
- Upgrade to 10 MW with a TRIGA-fueled core (Appendix A)
- Upgrade to 10 MW with a SILOE-type core (Appendix B)
- Decommission MITR-II (Section 3.9)

For each option, the key technical considerations were addressed and the costs were estimated. The following sections and appendices outline the results of these studies.

3 STUDIES RELEVANT TO RELICENSING AND UPGRADING

In this section, several studies relevant to relicensing and upgrading have been summarized. The results of some of these studies (Sections 3.4 through 3.7) can be found in Sections 4 and 5.

3.1 Need for Relicensing and Upgrading

The MIT Research Reactor's (MITR-II) license, No. R-37, which was originally issued by the Atomic Energy Commission and subsequently amended by the US Nuclear Regulatory Commission (NRC), expires at midnight May 7, 1996. Operation of the reactor beyond that date and time will require the issuance by NRC of either a license extension or a new license. The options being investigated by MIT include:

- a) An application has been submitted for an extension to recover time during which the license was in effect but the reactor was under construction, shutdown for modifications, or engaged in low-power testing. This option would provide almost five more years of operations. MIT approved the request and it was submitted to the NRC on March 31, 1994.
- b) Reviews and analyses have been made related to relicensing the facility at its existing power of 5 MW or at some higher power. The maximum practical power level is 10 MW because that is the cutoff between a research and a test reactor and test reactor status would lead to significant increases in licensing cost and costs for required infrastructure.

Relicensing is both a challenge and an opportunity. It is the former because of the need to assemble material that documents compliance with all applicable regulations. It is the latter because of the possibility of upgrading the facility and assuring a viable research reactor for a few more decades.

3.1.1 Upgrading Required to Maintain a Competitive Facility

The MIT Research Reactor was one of the first university research reactors to be built in the United States. From the time of its initial criticality to the early 1970s, the number of research and test reactors grew steadily, eventually reaching about 100 facilities. At present, there are about 50 with 36 of them operated by universities. The smaller facilities continue to cease operation at the rate of one to two per year. It is clear that if a facility is to attract first-rate research projects, then it must provide a competitive neutron flux. Therefore, consideration should be given to a power upgrade to the maximum allowed for a research reactor, namely 10 MW. This is the major upgrade considered for the proposed relicensing of the MIT Research Reactor. It entails performing reactor physics and thermal-hydraulic analyses of the proposed consequences of doubling power, installing new equipment (such as heat exchangers) as needed, and

revising or, in some cases such as the emergency plan, rewriting all of the supporting procedures and documentation.

3.1.2 Timing of the Relicensing Effort

Under current federal law, MIT is required to submit its request for relicensing and/or upgrade of the facility at least thirty (30) days in advance of the license expiration. Thus, all material must be received (not postmarked) on or before April 7, 1996. If this is done, then NRC will consider the renewal to be 'timely' and MIT will be allowed to continue operation at 5 MW while NRC reviews the material. Otherwise, our license will be voided. It will take NRC much more than 30 days for their review and, provided that the submission was judged to be 'timely,' MIT will be allowed to continue operation of the reactor during that review. This could conceivably result in several years of additional operation at the exiting power level. Conversely, if the goal is for an increase to a higher power level as soon as possible, then the application should be filed much earlier. No increase in power would be allowed until NRC completes its review.

An issue related to the possible upgrade of the reactor so that it can be operated at a higher power level is whether or not a 'construction permit' is needed. For renewal at 5 MW, one is definitely not needed. For an upgrade, one might be needed depending on the extent of the changes.

3.2 User Requirements

An important part of the design review process is the establishment of the users' requirements for the MITR-III research reactor facility. One of the main reasons for an upgrade of the existing MITR-II is to provide new, improved research facilities. Thus, it is important to develop an accurate, comprehensive and realistic compilation of what types of facilities and what types of operational characteristics might be required or desired of a reactor to be used in a research environment for the next 10 to 20 years.

Basically, the determination of user requirements and wishes has been carried out by interviewing a variety of research scientists regarding the types of facilities and reactor performance they would require, as well as what they could envision as being required in the future. From this extensive interview process, a distilled listing of user requirements was created. This listing follows, indicating the type of research involved and the characteristics of the research facility.

3.2.1 Neutron Beam Tube for Scattering and Interaction Experiments

- Peak thermal neutron flux greater than 1×10^{14} n/(cm²-sec); the higher the better.
- Tangential beam tubes
- A Cold Neutron Source

3.2.2 Neutron Activation Analysis and Isotope Production

- Peak thermal flux greater than 5×10^{13} n/(cm²-sec)
- Fast transit facilities for study of short-lived isotopes
- Flexibility for the installation of a cryogenic system
- Minimum flux gradients
- A high flux "wet" facility (e.g., D₂O cooled) as large as 6 inches O.D.

3.2.3 Medical Therapy Beams

- Beam characteristics at beam outlets should be variable, ranging from a high quality epithermal flux of 1×10^9 to 1×10^{10} n/(cm²-sec), to a high quality thermal flux of approximately 1×10^{14} n/(cm²-sec)
- Easy access to medical beam shutters

- Minimum distance from core peak flux region to patient
- Maximum solid angle of beam available at target
- Consideration of both horizontal and vertical medical therapy beams
- Collimated beams

3.2.4 In-Core Irradiation Facilities

- 1.5 to 2 times higher fast flux than currently possible in the MITR-II ($E > 0.1$ MeV)
- Space for three or four in-core facilities
- Space for in-core positions of greater than 2 inches in diameter
- Flexible operating schedule to accommodate experiments

3.2.5 Digital Display and Control Capabilities

- Control room layout and instrumentation suitable for implementation of various digital control studies
- Flexible operating schedule to allow experiments on transient response and control experiments

3.2.6 Operational Requirements

- Use the existing containment structure reactor block and core
- Simplify routine operational and emergency procedures

To summarize the above user requirements, it is instructive to examine how these requirements differ from the research facilities provided with the current MITR-II. Compared to the MITR-II, nearly all users now desire higher flux levels, greater beam quality, and more flexibility in controlling beam characteristics. Thermal columns are no longer viewed as essential research facilities, while boron-neutron capture therapy (BNCT) and medical therapy facilities are now emphasized. Facilities to allow experimentation with advanced digital and diagnostic equipment are also needed for a future reactor. Beams for neutron scattering should have the highest possible flux.

3.3 Requirements for Relicensing

Even if the MITR-II were to be relicensed at its existing power level of 5 MW, certain upgrading would be required for a reliably functioning reactor; safety is not an issue. This would involve both systems and procedures. The material that follows is not intended as a comprehensive list. Rather, the objective is to give a feeling for the types of activities that are needed. Regarding physical systems, the following sections describe the status of important MITR components:

3.3.1 Reactor Core Tank

The MITR-II lower core tank is the reactor structural component which is exposed to the maximum neutron fluence. An evaluation has been made as to the effect of long term neutron irradiation on core tank material properties, principally ductility reduction [1]. This evaluation has shown core tank material properties to be within acceptable limits even after 20 additional years of operation at 10 MW. If possible, actual metal samples should be taken and evaluated for ductility.

The core tank has also been analyzed for possible corrosion. Analyses have shown that no significant corrosion should be present even in the lower core tank after extended 10 MW operation. Periodic visual examination of accessible surfaces has determined that no significant corrosion has occurred to date.

3.3.2 Reactor Beam Ports and D₂O Reentrant Thimbles

The MITR-II currently has twelve beam ports with associated reentrant thimbles which extend into the D₂O reflector tank to a high flux region adjacent to the reactor core. There is a possibility that long term exposure to high humidity and heat in a radiation environment may have caused some degradation of the D₂O reentrant thimbles. The problem is that, under such conditions, nitric acid may form if there is any air present. This is believed unlikely because the thimbles are kept under a cover gas [2]. Further analysis and possible visual examination is desirable.

In addition, five of these beam ports have lead shutters which have been permanently affixed in the open position because of swelling concerns. It is desired to redesign and reconfigure these shutters so that they can be reliably operated in the future.

3.3.3 Outer D₂O Reflector Tank

Because of exposure to temperature, radiation and humidity, it is possible that the outer surface of the aluminum D₂O reflector tank has experienced some corrosion, particularly where the D₂O reentrant thimbles are joined. Several inspections have been made, the last in 1989. These confirmed the absence of major corrosion. However, some white scale (probably aluminum oxide) was noted. Analyses to date indicate that the tank

is expected to be satisfactory for relicensing [2]. Additional inspections will be made periodically.

3.3.4 Core Shroud Ring

The top section of the MITR-II core shroud is sealed to the core tank by three rings of polyethylene packing. The condition of this sealing ring is unknown, and there could have been considerable degradation over almost twenty years of reactor operation. This seal probably needs replacement. However, assessment has been made and negligible bypass flow was measured [2].

3.3.5 Cooling Towers

The current wooden cooling towers are 35 years old. Severe weather exposure, primarily ice buildup in the winter, has caused deterioration of the structural components and also heat removal capability. Sections of the flooring on top of the tower are becoming unsafe and concern is growing about other components.

The cooling tower fans and associated motors and crankcases were last refurbished in 1973 and need repair. The fans remain serviceable but the motors and particularly the crankcases need immediate replacement. The cooling towers must be replaced for continued operation beyond 1996.

3.3.6 Underground Secondary Piping

The secondary system transports heat from the secondary side of the main heat exchangers to the cooling towers. The portion of the system that is above ground (valves, cooling tower riser pipes, etc.) was replaced in 1989. The portion that is underground will need replacement within the next few years.

3.3.7 Graphite

The graphite reflector surrounding the reactor has, over a number of years, been exposed to moisture. However, periodic inspections have shown no evidence of any associated degradation.

3.3.8 Primary Heat Exchangers

The three primary heat exchangers are barely adequate for 5 MW and during hot, humid weather, reactor power must be reduced. Also, a number of tubes have been plugged in one of the existing units. This heat exchangers must be replaced for continued operation.

3.3.9 System Pumps

Six major pumps are used in the primary, secondary, D₂O, and shield systems. These date from the 1960s (or earlier) and no spare parts are commercially available. Most, if not all, of these should be replaced for continued or upgraded operation to assure reliable reactor operation.

3.3.10 Shield and D₂O Heat Exchangers

The shield heat exchanger, because of its design, requires replacement every few years. A redesign might be in order to minimize maintenance. The D₂O heat exchanger is currently adequate for continued 5 MW operation, but would require replacement for 10 MW operation. Periodic assessments of both heat exchangers should be made.

3.3.11 Flow Venturi and Orifice Plates

Monitoring of system flows is made with orifice plates (for secondary, D₂O, and shield systems) and a Venturi meter (for primary). These are currently adequate, but a long term assessment will be made to determine if these plates have been eroded.

3.3.12 Control Blade Drives

The current control blade drive design is adequate with the exception of the drive motors. The seven existing motors have no commercially available spare parts, so maintenance is difficult with parts having to be made in-house. Therefore the design must be modified to accommodate currently available motors.

3.3.13 Control Blade Magnet Designs

The six present control blade magnets, which provide for an emergency shutdown of the reactor, are exposed to radiation, heat and humidity which causes degradation. As a result, these magnets must be replaced every few years. A redesign of the magnets would save a significant amount of labor since the current magnets are manufactured in-house.

3.3.14 Control Blades

The six Boron-stainless steel control blades are presently replaced every 5-6 years because of boron burn-up. This replacement is labor intensive and costly as well as being a waste disposal concern. It is desired to redesign the control blades using Hafnium which would significantly lengthen blade lifetime.

3.3.15 Primary Ion Columns

As part of the primary cleanup system, the ion columns contain a mixed bed resin which must be replaced 3 to 4 times per year. The current design, although adequate, should be modified to improve ease of resin exchange, resulting in decreased labor and personnel radiation exposure.

3.3.16 Containment Building Shell

The steel containment building shell is protected from below grade corrosion by a cathodic protection system. Underground anodes to this system were replaced in 1992 and should be adequate for a number of years. Building integrity is checked annually.

3.3.17 Airlock Gasket Design

Both of the MITR-II personnel airlocks utilize inflatable rubber gaskets for sealing the doors. Occasionally these one-of-a-kind gaskets must be replaced at considerable expense and labor expenditure. A redesigned simpler door and gasket would ease these concerns.

3.3.18 Truck Airlock

The facility was built with an air lock large enough to permit the entry of large vehicles. The intended use was for ambulances transporting patients who would receive neutron capture therapy and for trucks carrying large shielded casks for spent fuel. The lock has three gaskets, one on each door and an expansion joint. Those for the outer door and the expansion joint need replacement. Also, the seating surface for the outer door gasket needs extensive repair. Finally, the structure of portions of the lock itself requires extensive repair because of exposure to the weather over 35 years.

3.3.19 Support Structures

Because of minor piping leaks over thirty years, there is considerable corrosion buildup around the reactor upper shield. The upper shield access ring, upper shield ring, and upper and lower annular rings should be removed and refurbished prior to continued or upgraded operations.

3.3.20 Emergency Batteries

Sixty batteries in a bank serve as emergency power in case of a loss of on-site electrical power. Over a period of thirty years, the capacity of these batteries has been significantly reduced. Replacement is required for continued operation beyond 1996.

3.3.21 Emergency Core Cooling System

The present ECCS consists of two nozzles through which water can be sprayed down on the core upon loss of coolant. The spray distribution is tested semiannually and must be extensively reevaluated whenever new in-core experiments are proposed. A simpler system is desired in which an even cooling distribution can be easily verified. (Note: This system is not essential to reactor safety but gives added defense-in-depth.)

3.3.22 System Valves

A number of valves in various coolant systems require special operation, sometimes in high radiation areas, in order to minimize temperature gradients for sensitive experiments and reactor fuel. Although these valves are currently adequate, it is desired to replace them with remotely operated valves to facilitate ease of operation.

3.3.23 Motor Control Centers

There are two motor control centers plus five major breakers. The insulation on many of these has aged to the point where replacement is necessary.

In addition to the upgrade of the MITR-II's systems, its procedures will require review and in some cases revision. For example, the US Nuclear Regulatory Commission feels that our emergency plan is too conservative and they have indicated that it must be revised prior to relicensing. Also, the training program (which is excellent) no longer completely conforms to NRC guidelines.

3.4 Steady-State Analysis to Upgrade the MITR to 10 MW Using Uranium Silicide Fuel

3.4.1 Introduction

This research focused on refueling the MITR-II with a different fuel as part of the studies to raise the operating power level and the neutron flux. This would be performed without any major modifications to the present reactor design including the thermal hydraulic design of the MITR plate type fuel element or decrease in the fuel cycle length.

One proposed fuel type is highly enriched uranium silicide dispersion (U_3Si_2 -Al) fuel. This fuel was developed for the Reduced Enrichment Research and Test Reactor (RERTR) Program to satisfy the US Department of Energy low enrichment safeguards for University Research Reactors [3]. Due to its higher uranium density of about 4.8×10^{-3} kg-U/m³, as compared to 1.59×10^{-3} kg-U/m³ in the present aluminide fuel, a higher operating power level can be obtained while keeping the same fuel cycle length as in the present reactor. Furthermore, this fuel has a higher melting temperature and thermal conductivity than the currently used fuel (uranium aluminide).

Another proposed fuel type "ELAF" fuel is also being studied as a possible MITR fuel for operation at 10 MW. The ELAF fuel is a high density uranium aluminide fuel with Boron as a burnable poison. This ongoing study is summarized in Appendix E.

3.4.2 Objective

The objective of this research is to evaluate the measures that can be taken to achieve an operating power level of 10 MW, which is twice the current MITR-II rating. Specifically, this research examined the neutronic and thermal hydraulic parameters necessary to determine if the MITR-II can be refueled and operated with uranium silicide dispersion fuel to meet the present MITR-II limiting safety system setting condition at a power level of 10 MW.

The objective was subject to the satisfaction of three design constraints on the thermal hydraulic characteristics of the reactor. The first is that no boiling shall occur anywhere in the core during normal operation of the reactor, this is currently satisfied with a significant margin. The second is that the present primary piping is to be used with larger pumps to minimize upgrade costs. Finally, the fission product decay heat must be removed by natural convective cooling during a loss of flow transient [4]. The power and core temperature distributions were evaluated analytically and numerically to assess the limiting operating conditions for the proposed core based on these three constraints. The results of the research incorporating uranium silicide fuel are summarized in Section 4.1.

3.5 MITR Secondary System Design Analysis

The purpose of the MITR secondary system design analysis project is to perform analysis required for properly selecting the necessary components of the secondary cooling system for the MITR-III Relicensing project.

The objectives of this project are:

1. To design a secondary system capable of exchanging heat for the 10-MW upgraded MITR-III.
2. To design a secondary system capable of exchanging heat for the 5-MW relicensed MITR-III.
3. To look at options of upgrading from 5-MW to 10-MW at a later date.

The third objective requires that the design for the 5-MW system will allow future procurement and installation of additional components to accommodate future power upgrading possibility.

Sections 4.2 and 5.2 summarize the different models selected to replace the existing secondary system.

3.6 Computer Simulation for Transient Analysis of the MITR

A computer program TAMLOC (Transient Analysis of MITR LOP Components) has been developed to simulate several transients for the MIT Nuclear Research Reactor. The transients under consideration include slow to moderate reactivity insertions, loss of heat sink, and pump coast-down with attendant natural circulation.

The program was written in FORTRAN and based on the adaptation of the computer code SPK [5] to the MIT Reactor system. The SPK code was originally developed as an analytical model for supporting on-line operator assist systems in PWRs.

To demonstrate that TAMLOC's dynamic models can realistically simulate the MITR behavior, simulated results were compared to available data for shutdown and startup transients as well as a pump coast-down transient. Overall, it has been demonstrated that TAMLOC can adequately simulate the MITR responses and provide reliable information for the range of conditions studied.

The modeling of the reactor system is summarized in Appendix C. Any detailed additional information on the modeling can be found in Trosman's Thesis [6]. The results of the simulations at 5 MW and 10 MW are given in Sections 5.3 and 4.3, respectively.

3.7 Preliminary Cost Analysis of Relicense and Upgrade

3.7.1 Introduction

The current MIT Reactor, MITR-II, has been in operation since 1975. Several of the reactor's key systems and components were installed during or prior to the 1975 upgrade. Some of these pieces of equipment are reaching the end of their useful lives as performance deteriorates, maintenance becomes more frequent, and spare parts become increasingly difficult to obtain. This cost analysis is based on a complete review and preliminary cost estimate reported in June, 1993 [7]. Updated values of component costs have been included here, and updated system requirements are reflected in these new costs.

3.7.2 Relicensing at 5 MW

This analysis was obtained by assuming that a relicensing of the current MITR-II at 5 MW will lead to another 20 years of operation. Therefore, any component that will need to be replaced in the next 20 years is included in this estimate.

In most cases it would make economic sense to replace these aging components at the beginning of the new licensing period. The reasons for this are threefold:

- 1) The new components can be installed during the same shut-down period. This will avoid the lost revenues associated with multiple extended shut-downs.
- 2) The new components are generally more efficient and will require less maintenance than the older models that they are replacing. Replacing aging components will reduce maintenance downtime and provide higher reliability of operation.
- 3) The new components (particularly heat exchangers and pumps) can be modeled and analyzed using the methods outlined in other sections of this document. The SAR and Tech Specs can be modified to reflect the performance of this updated system.

The reactor will need to be shut down for several weeks to perform necessary replacements and maintenance on the secondary side. This down time should be used to install and test the new primary side and other components to avoid a later shut-down. The total cost for the relicensing is sensitive to the estimated down time to perform all needed upgrades. For this analysis a down time of three months was estimated. This downtime can be spread over a period of months or years. The results of this study are summarized in Section 5.4.

3.7.3 Upgrade to 10 MW

Several key components in the heat removal system will need to be replaced with larger units to accommodate the increased heat generation rate associated with 10 MW operation. This analysis was obtained by assuming that the 10 MW successor to MITR-II will operate for at least 20 years. Therefore, any component that will need to be replaced in the next 20 years is included in this estimate.

The reactor will need to be shut down for several weeks to perform necessary upgrades on the secondary side. This down time should be used to install and test the primary side and other components to avoid a later shut-down. The total cost for the upgrade is sensitive to the estimated down time to perform all needed work. For this analysis a down time of four months was estimated. These four months of down-time will not occur all at once, but will be spread over a period of months or years. The results of this study are summarized in Section 4.4.

3.7.4 Structure of the Cost Analysis

Engineering and Design Costs

Engineering and Design encompasses all of the research and development activities that will be undertaken to meet NRC requirements and to arrive at a set of operating parameters for the reactor with its new components. Much of this work will be done in the form of research projects to be completed by S.M. and Ph.D. candidates at MIT over the next several years. Engineering and Design also includes the time and resources that must be devoted to preparing the required documents for relicensing.

Operating Costs

Operating Costs include costs incurred by the relicensing project that do not involve the purchase and installation of equipment, or the planning and design of the system. A large portion of the Operating Costs are lost income and payroll costs. During the period when the reactor is down for installation of new components, the Nuclear Reactor Laboratory (NRL) will employ its regular staff of operators, engineers, technicians, skilled craftsmen, and administrative personnel to work on the relicensing effort. In addition, it is likely that it may be cost-effective for the NRL to hire supplementary technical staff. Operating Costs also include normal material expenditures, and income lost due to shutdown.

Other Direct Costs

Other Direct Costs are the costs necessary for the purchase and installation of new components. This category also covers the refurbishment, decontamination, or disposal of components, where necessary. These costs are arrived at by making a take-off list of all components and tasks that will be needed to complete a given operation. Some tasks,

such as replacing a section of pipe, are well-defined and can be easily broken down into their constituent subtasks, yielding an accurate estimate. Others are very roughly outlined because insufficient data is available at this time to break down the task.

Direct costs include both materials and services. Labor service costs are found by estimating the man-hours required for a given task and multiplying by the appropriate labor rate for labor in the Boston area. Material costs are the results of estimates from suppliers or listings in construction material catalogs. The cost of a component includes the contractor's labor costs, installation, and shipping.

The costs of removal and disposal are included as a separate direct cost category. Some components will require decontamination and/or special disposal. The future of low-level radioactive waste disposal is uncertain at the current time. Therefore, the disposal costs for radioactive components only includes steps taken to remove and pack the component for shipping to a disposal site. The volume of radioactive waste which is generated in a relicensing for 5 or 10 MW, is expected to be sufficiently small that it could be stored on site indefinitely or until a low level waste repository is available in Massachusetts.

3.8 Minimum Requirements to Relicense

The relicensing of nuclear reactors, either those for commercial power or those for research, is challenging both because of the work involved and because the governing regulations are not yet finalized. At present, it appears that the US Nuclear Regulatory Commission has three objectives in any relicensing:

- (a) Verification that the material condition (i.e., integrity of core housing, vessel, and containment building and viability of wiring, electrical components, and major equipment such as pumps and heat exchangers) of the facility will be adequate for the requested duration of the renewed license.
- (b) Assurance that the facility is in compliance with current regulatory practices. That is, equipment designed decades before a given regulation was enacted must be modified to meet current requirements. The same applies to procedures.
- (c) Assurance that all supporting documents (i.e., safety analysis report, design drawings, etc.) reflect the actual as-built status of the plant. In other words, have all the modifications, particularly instrument changes, been picked up in the documentation?

These objectives translate to a major administrative burden. This is discussed below relative to the three most likely options for relicensing of the MIT Research Reactor.

The first relicensing option has been initiated by application to the NRC a request for an extension to the current license to recover time spent in construction. Reactor operating licenses date from the day that construction is authorized. It would be more logical to date them from the completion of construction. NRC is amenable to this argument. Such an extension might add almost five years to the reactor's current R-37 license.

An interim license extension has certain advantages. It buys almost five years and the radioactive waste disposal issue, which is very muddled now, might be moving toward resolution by 2000 or 2001. Also, funding for university reactors, which is currently inadequate, would probably be rising in 2000 because the impact of losing some of these facilities would have taken effect. The disadvantage is that MIT risks exposure to a public hearing on the license renewal on two separate occasions, once in 1996 for the interim license extension and once in 2001 for the long-term license renewal. This is a major risk because once a dedicated group obtains intervenor status they can use the NRC legal system to waste enormous amounts of Senior Staff and legal time and thereby damage morale and drive up costs.

MIT approved submission of the license extension request and it was submitted on March 31, 1994. Thus far, NRC has not acted on the request.

The second option is to relicense the reactor 'as is,' that is at the existing 5-MW power level. It is this option, relicensing at 5 MW, that is the absolute minimum that is needed. The following would be required:

- (i) An assessment of the materiel condition of the facility with emphasis on the items that are difficult to inspect but important to safety. These include the beam tube reentrant thimbles in the D₂O, the gasket on the lower annular ring, cooling coils for the shield system, flow venturis and orifice plates, the fixed poison plates in the upper part of the core, the grid latch, circulation valves, core housing, core vessel, D₂O vessel, primary and D₂O piping (particularly in the pipe tunnel where some vibration may exist), heat exchangers, cable runs, the emergency power battery bank (needs replacement), and the containment building. In addition, equipment not essential to safety but important to operation should be assessed, especially the underground section of the secondary piping and the cooling tower.
- (ii) A review of reactor systems and procedures for compliance with NRC regulation effective as of 1996. Some areas of weakness are as follows:
 - (a) As noted earlier, NRC considers our emergency plan to be too conservative. Revision were prepared on January 1994. Further revisions are on progress.
 - (b) The reactor's training program, which was authorized under a Commission letter in 1979, does not conform to current NRC guidelines. (Note: It is an excellent program; the issue is not one of content but of the correct language, etc.)
 - (c) Documentation on the design basis of the reactor's nuclear safety system was lost shortly after the 1973-1974 modification. A new system was ordered with DOE funds in July 1994.
- (iii) A review of the reactor's supporting documentation. The first step is to have licensed personnel trace out all mechanical and electrical systems and compare results against the design drawings. The NRL has a very active quality assurance program and drawings are updated as changes are made to equipment. Nevertheless, some differences undoubtedly exist, most probably in the electrical area. Once the drawings are verified to be accurate, the Safety Analysis Report (SAR) should be updated. The SAR is a 'living' document and has been updated regularly. However, there has never been a complete rewrite. A further difficulty is that many of the safety analyses reference documents that were current in the 1950s and early 1960s. Most of these analyses could benefit from a reassessment done with current technical data and modern computing technology. Upon issuance of a new SAR, other derivative documents will then require updating including:
 - The Technical Specifications
 - The Emergency Plan
 - The Training Manuals

The above is a major undertaking with several man years of Senior Staff time being required.

- (iv) A study should be undertaken to simplify operation of the reactor. This is not an NRC objective in relicensing. However, NRC has indicated that they would be amenable to suggestions in this area. At present, the reactor is too complex in that too many tests and calibrations are needed and there are unnecessary safety features. This was fine when the regulatory burden was lower. However, that burden is growing rapidly and the result is that much of the Senior Staff's time is spent on administrative issues that do not contribute to safety. For example, the reactor has several hundred procedures, many for instrument calibrations. Any change in any one of these procedures requires a formal safety review that in turn requires documentation of negative findings (i.e., that no unreviewed safety question exists). The net result is an enormous waste of effort on issues that do not contribute to safety. The only solution is to eliminate or simplify as many procedures as possible. Examples include:

- Elimination of scrams on secondary flow. These are not required by NRC.
- Elimination of redundant shield system flow scrams.
- Minimization of pressure gauge calibrations.
- Modification of the containment building airlocks to use a less complex seal mechanism.
- Permission to have two empty positions in the core during refuelings. (This would cut the number of moves needed to invert a fuel element by two-thirds.)
- Simplification of the refueling calculations. We currently do full-core, 3-D calculations that require about one week of a Senior Staff person per refueling. We refuel about six times per year.

The third relicensing option is to upgrade the MIT Research Reactor to some higher power level, 10 MW. The NRC has indicated that MIT can either renew the existing license at the present power level or upgrade, but not both because of the cost involved. NRC places this cost at \$1 M and, hence, would not view with favor a license renewal at 5 MW followed in several years by a request to upgrade. An upgrade would require, as a minimum, all of the items listed above for relicensing at 5 MW plus significant additional theoretical analysis including:

- Thermal-hydraulic and reactor physics studies to support the upgrade.
- A review of the design basis accident and its consequences.
- Rewrite of the emergency plan.
- Revision of the reactor's refueling strategy. Fuel element usage is not a linear function of power. We need to be certain that, even at 10 MW, only a few refuelings will be required per year.
- Assessment of the existing biological shield and radioactive effluent release rates.

3.9 Information on Decommissioning

One of the options regarding the MIT Research Reactor would be to decommission it when its present license expires, either in May 1996 or, if an interim extension to recover construction time is obtained, in 2001. The US Nuclear Regulatory Commission requires that all licensed facilities maintain funds to cover the eventual cost of decommissioning. This in turn means that accurate estimates of those costs must be available. To that end, MIT retained the services of the General Electric Company (GE) in 1988. The GE study, copies of which are on file at the NRL, placed the cost of decommissioning the MITR-II at \$9 M in 1988 dollars [8]. There are two principal factors that contribute to this cost; labor (80%) and radioactive waste disposal (20%). The NRL periodically reviews these two factors in order to maintain the total cost as accurate as possible. Labor costs have either held constant or declined slightly since 1988. However, waste disposal costs have risen sharply as the result of the federal and state governments' continued inability to develop sites for the disposal of low-level radioactive material. After discussion with the MIT Radiation Protection Officer and also with agents for waste disposal contractors, we estimated the cost of decommissioning the MITR-II at \$14.0 M and \$20.0 M in 1996 dollars [9] (approximately between \$12.0 M and \$17.0 in 1993 dollars).

The decommissioning option is not as simple as the dollars-and-cents estimate given above might make it appear. In fact, the complete decommissioning and decontamination of the facility is, for all practical purposes, likely to be impossible in the foreseeable future. The problems are as follows:

- (a) The first step in the decommissioning process would be to return the spent or partially-spent reactor fuel to its owner, the US Department of Energy (DOE). This requires the availability of a shielded cask that is approved by NRC. At present, there is only one such cask in the United States and its license may not be renewed. (Note: In the 1980s, the DOE spent several million dollars attempting to refurbish and license two other casks. However, the effort was abandoned when it became evident that these casks would not be licensed by NRC.) Hence, it is not clear that we would be able to ship the fuel to DOE in 1996.
- (b) Even if there is a cask available for use in returning the spent fuel to DOE, DOE may not be able to accept it. In September 1992, DOE decided to halt the reprocessing of spent university reactor fuel. (Fuel from commercial nuclear plants is 2-3% enriched and has not been reprocessed for decades. But most university reactor fuel was 93% enriched and has always been reprocessed.) The reasons for DOE's decision were (1) the availability of large amounts of Russian enriched uranium and (2) the large volume of low level waste produced during reprocessing. DOE's plan is now to store URR fuel on-site in pools at Savannah River until an interim dry storage facility can be built. The fuel would then be kept in that facility until a permanent high-level waste depository becomes available. It will take at least four years to build the interim storage facility. The problem for universities is that Savannah River lacks the needed storage space in its pools and once the existing

spaces are filled, universities will have to hold their spent fuel on site. In the first half of 1993, MIT managed to return 48 spent elements to DOE. Another eight were returned in 1994. However, no further returns will be accepted until the interim storage facility comes on line. At the earliest, this would be late in 1998. Quite probably, the actual date would be much later.

- (c) The obvious solution to the problems identified in paragraphs (a) and (b) above is to disassemble the core and store the spent fuel on site. However, this also creates problems. First, a third of the NRL's capacity to store spent fuel is located in the core tank itself. Second, spent fuel is required to be kept 'self-protecting' in the sense that its radiation level is such that anyone trying to steal it would die in the attempt. Spent fuel, even if in long-term storage, can be kept above the self-protection limit by reirradiating it for a few days every few years. However, if the reactor were to be shut down, this would not be possible and radiation levels would, after about a decade, drop below the self-protection limit. Federal law then requires that armed guards be on site to protect the storage location continuously. This can become very expensive. Finally, as long as the fuel is present, NRC-licensed fuel handlers would be required. These people would not have to be as well trained as a reactor operator, but there would have to be a formal training program for them. Also required would be a quality assurance program, a security plan with alarm systems, and a calibration program for the instruments used to monitor the fuel.
- (d) For decades, the only locations available for the disposal of low-level radioactive waste were Beatty NV, Hanford WA, and Barnwell SC. None of these has unlimited capacity. In the late 1980s, the federal government acted to remedy this situation by requiring each state to join a compact (presumably of adjacent states) and to build a waste disposal site for the collective use of all members of the compact. In order to provide incentive, the federal law established a timetable for loss of access to the three existing sites. At present, Massachusetts (and most other states) has lost access in July 1994 to all out of state disposal locations. Thus, it will not be possible to ship out low-level radioactive material until Massachusetts opens its own disposal facility. This appears to be at least a decade away. This means that the core housing, reactor vessel, primary piping, and biological shield (all of which have significant induced radioactivity) would remain on site and require periodic monitoring by the MIT Radiation Protection Office for many years. Also, the reactor site would, of course, not be available for alternate usage. Finally, it is likely that when disposal sites do become available, the cost per cubic foot of waste will be substantially greater, perhaps by an order of magnitude.

In summary, the cost of decommissioning is rising rapidly because of increases in the cost of disposal of radioactive waste. Also, complete decommissioning and decontamination is not a realistic option given the present difficulties in disposing of both high- and low-level radioactive material. Finally, it should be understood that as long as the fuel is on site, significant resources will be required for its surveillance and handling. These include two-three people licensed to handle the fuel, radiation protection people, and depending on the duration of the storage interval, around-the-clock security guards.

(Note: The fuel handlers would be needed for only a few hours per week except during actual shipments. However, they must be licensed on the MIT facility and licenses are not transferable. So, temporaries cannot be hired or borrowed from another reactor. The required expertise is considerable and it is difficult to envision obtaining part-time workers for this purpose. Perhaps they could be part of the MIT Radiation Protection Group.

4 SUMMARIES OF RESULTS APPLICABLE TO AN UPGRADED 10 MW MITR

This section contains the results of studies applicable to upgrade the MITR to 10 MW. These studies and their objectives were introduced in Section 3.

4.1 Results of Steady-State Analysis to Upgrade the MITR to 10 MW Using Uranium Silicide Fuel

The following are results obtained in the study of upgrading the MITR to 10 MW using highly enriched Uranium Silicide dispersion (U_3Si_2 -Al) fuel.

4.1.1 Neutronic Conclusions (Steady-State)

The neutronic analyses were made with the MCNP Monte-Carlo code [10]. These analyses were based on core configuration Core II, which corresponded to one of the initial core loadings for the MITR-II. It consisted of 22 fuel elements and 5 dummy assemblies arranged as shown in Figure 4.1-1 without stationary absorber inserts present. This core configuration was chosen because it has the least number of fuel elements and no element is adjacent to two stationary absorber insert locations. This core configuration was employed to calculate the maximum uranium loading that could be inserted into the core and still satisfy the Technical Specification requirement for shutdown reactivity margin, i.e. making the reactor sub-critical with the most reactive control blade and the regulating rod fully withdrawn plus 1% $\Delta k/k$. It was found that the maximum uranium loading that would meet the present Technical Specification shutdown reactivity margin without burnable poison was 2.6×10^3 kg U/m³ of fuel "meat" utilizing highly enriched (93 % U^{235}) uranium.

The fuel plates with the maximum peak power and producing the maximum power for the analyzed core are shown in Figure 4.1-2. The fuel plate having the maximum peak power occurred in fuel element position C-1. This peaking occurs because the control blades are located right next to the fuel plate, causing the power to be depressed in the upper half of the plate, while the light water in the downcomer causes a large peak power at the entrance of the channel. This plate had a radial peaking factor of 1.131 and an axial peaking factor of 2.475. The fuel plate producing the maximum power was located in fuel element position A-1. This fuel plate had a radial peaking factor of 1.388 and an axial peaking factor of 1.670. Even though it had a lower peak power than the maximum peak power fuel plate, i.e. the C-1 element, the limiting condition for prevention of the onset of nucleate boiling was reached in this fuel plate. As the power increases, it is predicted that incipient boiling will begin first at the top of the fuel plate. Finally, the onset of nucleate boiling is predicted to occur in the middle of this fuel plate after both ends have been depleted due to burnup.

Figure 4.1-1. Core Configuration Analyzed Using Uranium Silicide Dispersion Fuel

Figure 4.1-2. Power Distributions for the Analyzed Core

A complete analysis of the expected in-core and ex-core fluxes available for experimentation was not performed. Nevertheless, some insights gained from this research are presented and compared to the present MITR-II. Since the power has doubled to 10 MW, the fast flux will double. The thermal in-core flux will, however, only increase by a factor of approximately 1.2 because the uranium loading has also increased by 1.6 over the present fuel loading. As for the thermal fluxes in the beam ports and medical facility, they should both double because they depend on the thermalization of the fast flux, which has doubled. As a result of these flux increases, an analysis of the increased activation of the core, in-core experiments, and surrounding material should be performed together with an assessment of the required increased radiation shielding. Preliminary assessments of increased shielding requirements indicate that 10 MW operation is manageable since the major issues are radiation levels near penetrations in the main biological shield rather than general leakage through this shield.

Finally, the same fuel management strategy presently employed can continue to be used. The only differences between the present and proposed cores is the higher uranium loading which raises the core reactivity. Therefore, the proposed core will not be able to accommodate 27 fresh fuel elements at any one time. However, this creates no problems because better flow distribution is achieved when at least one dummy assembly or experimental facility is located in every azimuthal section of the core. Finally, it has been calculated that the proposed core will have a slightly longer fuel cycle length than the present core.

4.1.2 Thermal Hydraulic Conclusions (Steady-State and Preliminary Transient Analysis)

Thermal hydraulic calculations were performed to investigate the advantages of utilizing fins. Computer calculations using the FLUENT code [11] have confirmed that the surface effectiveness is 1.9, as previously calculated and measured [12]. Also, calculations show that if corrosion should build up on the finned surface, the heat transfer from the clad to the coolant will not be significantly degraded and should not cause any problems for the fuel plates to achieve a higher burnup. Furthermore, thermal conduction calculations through the bottom and top of the fuel plates were performed to investigate how much heat was leaving through these regions. It was found that the predicted peak heat fluxes at the top and bottom of the fuel plates decreased by greater than 30 % in the first and last 10 millimeters of the fuel regions. Finally, entrance effects were taken into consideration due to the development of a boundary layer as the coolant first enters the heated flow channel. This should lower the wall temperature at the entrance of the heated flow channels and thus, increase the margins for incipient boiling.

The present limiting safety system setting for preventing the onset of nucleate boiling was maintained. Based on calculations using PARET [13], as long as the reactor operates within the parameters given in Table 4.1-1, no incipient boiling is predicted to occur. The safety limit of preventing the onset of flow excursion instability in the event that fully developed nucleate boiling should occur in the reactor was also defined. If the power increases with the average core outlet temperature remaining constant at 63°C and a flow

rate of 113.5 kg/s, the onset of flow excursion instability will begin at 18.0 MW. On the other hand, if the power level remains constant at 10 MW with a flow rate of 113.5 kg/s, the average core outlet temperature can increase to a maximum temperature of 77.3°C before the onset of flow excursion instability begins (corresponding to an average core inlet temperature of 55.3°C). Finally, if the power and average core outlet temperature should start to increase past the limiting safety system setting defined in Table 4.1-1 with a core inlet temperature of 41°C, the onset of flow excursion instability will occur at 17.25 MW.

Table 4.1-1
Limiting Safety System Settings for the MITR-III

Number of Primary Pumps:	One	Two
Maximum Power, MW	5	10
Minimum Flow Rate, kg/s (gpm)	56.7 (906)	113.5 (1814)
Maximum Average Core Outlet Temperature, °C	63	63

Analyses of utilizing stationary absorber inserts with Uranium Silicide dispersion fuel were also performed. It was found that the power peaking at the bottom of the fuel elements caused by depressing the power in the top half of the core was too severe to prevent incipient boiling from occurring at 10 MW. The calculated maximum power at the minimum condition for flow rate and a core outlet temperature of 59.1°C with stationary absorber inserts present is 8.8 MW. The Safety Analysis Report for the MITR-II states that the maximum power with stationary absorber inserts and the same minimum conditions is 6.0 MW. This increase of 2.8 MW was achieved by performing a more accurate thermal hydraulic analysis, which included accounting for the entrance effects on the heat transfer coefficient, bottom and top thermal conduction, actual amount of fission energy deposition in the core, semi-statistical evaluation of the engineering hot channel factors, and lowering the inlet core temperature.

Table 4.1-2
Predicted Conditions for Onset of Nucleate Boiling*

	MITR-II: Safety Analysis Report		Thesis Calculations	
Power (MW)	6.0	8.8	9.25	10.8
Flow Rate, kg/s (gpm)	112.6 (1800)	112.6 (1800)	112.6 (1800)	112.6 (1800)
Average Core Outlet Temperature (°C)	62.2	59.1	66.7	63.0
Inlet Core Temperature (°C)	47.8	41.0	47.8	41.0
% of energy Produced in Core	95.0	96.67	96.67	96.67
Stationary Inserts Status	Present	Present	None	None

* See Section 4.3.2 for further analyses including transient response times.

Transient and accident analyses with and without scram were also performed on the proposed core. The transients included the loss of flow, loss of power and the maximum reactivity insertion, while the accident study consisted of evaluating a channel flow blockage. It was found that no incipient boiling will occur if the reactor scrams during a loss of flow or power transient. However, incipient boiling will occur following a loss of flow transient during natural convection, with a maximum clad temperature reaching 129.5°C, and this will cause no problems with decay heat removal. Preliminary reactivity induced transient studies indicate the maximum reactivity insertion sustainable without fuel damage is \$1.79 in 0.7 seconds if the reactor fails to scram. This is a higher rate of insertion than any foreseeable accident. Further studies on the use of the PARET code to predict the maximum allowable reactivity insertion is discussed in Appendix F. Finally, the consequences of channel blockage depend on how much of the channel is actually blocked. Even if a complete channel flow blockage occurs, no degradation of the cladding will occur as long as the reactor scrams.

4.1.4 Conclusion

From the viewpoint of the steady-state core neutronic and thermal hydraulic requirements, the proposed increase in the operating level to 10 MW using Uranium Silicide dispersion fuel is estimated to be feasible and achievable without any other modifications to the core. The maximum power that the reactor can safely remove depends on the occurrence of flow excursion instability, which is predicted to occur at a power level of 18.0 MW at 113.5 kg/s and average core outlet temperature of 63°C. The safety limit and limiting conditions for operation have been met for the proposed core. Further discussion of these limits and operating conditions are given in Section 4.3 where consideration of transient analyses are also included in the evaluation of the limits. This limiting safety system condition provides a conservative means of ensuring that the occurrence of flow excursion instability will not occur anywhere within the core, because if boiling is prevented, flow excursion cannot occur. In summary, the redesigned reactor has a geometrically identical core with an increased power density.

This estimate of the capability to upgrade in the operating power level was achieved by using the knowledge and experience gained from operating the MITR-II reactor, by evaluating more accurately the neutronic and thermal hydraulic characteristics for the present reactor without stationary absorber inserts, and by lowering the inlet core temperature. The MITR-II was originally conceived as a very conservative design. However, during the 38 years of operating the MIT Research Reactor, new and more accurate correlations have been developed, along with more experience in understanding the effects of finned surfaces on heat transfer. Therefore, a significant amount of unnecessary conservatism was eliminated in the present research, while still retaining a large margin of safety and confidence in the design and operation of the reactor.

By doubling the operating power level of the Massachusetts Institute of Technology Research Reactor, it can continue to meet the Institute's and the national research interest in nuclear science and technology applications in a wide range of fields.

4.2 Results of Secondary System Design for 10 MW Operation

The MITR-III operated at 10 MW would require a new cooling system to handle the increased power. To increase the cooling capability of the secondary system, several components of the present system will require replacement. Principally, the current heat exchangers, cooling towers, and pumps on the secondary system are not adequate for the increased power level. In this study, the replacement options for the heat exchangers and the cooling towers were investigated in detail. Further study is necessary to determine the secondary pumping capacity. Table 4.2-1 lists the secondary system design requirements to operate at 10 MW.

Table 4.2-1
Secondary System Specifications

Flow rate (gpm)	2500
Total Heat Load on Cooling towers (MW)	12.0
Max. HX Load, primary-to-secondary (MW)	11.0
Max. Load of one HX, primary-to secondary (MW)	5.5
Max. Aux. Load (AC, D20, Shield HX's) (MW)	1.0
D20 HX Load (MW)	0.4
Shield HX Load (MW)	0.4
Cold Leg Temp (°C)	27.0
Hot Leg Temp (°C)	45.0

Figure 4.2-1 shows a schematic of the secondary system designed to operate at 10 MW. Replacement and revision of the secondary system piping will be designed in more detail and with the applicable ASME code such as B31.1.

4.2.1 Heat Exchangers

There are three basic types of heat exchangers available: shell-and-tube, lamella, and plate-and-frame. The MITR-II presently has two shell-and-tube and one lamella heat exchanger. The lamella design is no longer being made for nuclear applications. The primary factors in choosing a heat exchanger design were the ability to fit into the available space in the equipment room, the chance of cross-talk between the primary and secondary sides, and the cost.

Figure 4.2-1. Secondary System Design (Dashed lines for 10-MW design)

Shell-and-Tube Heat Exchanger

In an earlier study, information on a suitable shell-and-tube heat exchanger for the MITR was obtained [14]. A quote was obtained for three shell-and-tube heat exchangers that would fit in the space envelope in the equipment room. The price for the heat exchangers varied from \$110,000 to \$175,000 each depending on what regulations the heat exchangers conform to and whether or not a nuclear "N" stamp was provided.

Plate-and-Frame Heat Exchangers

Initial information suggested that plate-and-frame heat exchangers would suit our purposes better. The National Bureau of Standards and Technology (NIST) is presently using plate-and-frame heat exchangers in their nuclear facility [15]. NIST was contacted to find out about their history and experience with the plate-and-frame heat exchangers. NIST presently has plate-and-frame heat exchangers on some of their auxiliary systems and will be replacing their primary heat exchangers with plate-and-frame heat exchangers some time next year. They have found the plate-and-frame heat exchangers to be quite reliable and no leakage problems were experienced. NIST also reported they have not had any problem with cross-talk. However, NIST pointed out that the plate-and-frame heat exchangers are more prone to fouling problems than shell-and-tube heat designs. Consequently, both primary and secondary sides must be kept relatively "clean." A better filtration system for the secondary system might be required if plate-and-frame heat exchangers are used. To offset the fouling problem, plate-and-frame heat exchangers are relatively easy to disassemble and clean.

With this positive report on the merits of the plate-and-frame design, Alfa-Laval was contacted to discuss the plate-and-frame design. Alfa-Laval described the plate-and-frame heat exchanger as a gasketed design which is more compact than the shell-and-tube design for equivalent uses. Plate-and-frame heat exchangers use mass-produced pressed plates, leading to less expensive capital costs. Thinner wall between sides and corrugations on the larger surface area of the plates contribute to better heat transfer and less material. On the negative side, they require more maintenance and compressed-fiber gasketing because of restrictions on the pressure that can be applied to the plate walls. A preliminary design and price quote was obtained from Alfa Laval. Table 4.2-2 shows the requirements given to Alfa Laval for the heat exchanger. Some of these requirements were changed later to reflect cooling tower requirements. Plate-and-frame heat exchangers that meet the MITR-III's requirements will cost \$14,000 each.

Table 4.2-2
Heat Exchanger Design Criteria

	Primary Side	Secondary Side
Fluid	Light Water	Light Water
Phase	Single Phase liquid	Single Phase Liquid
pH	5.5-6.0	7.0
Flow Rate	1100 gpm (69.4 kg/s)	870 gpm (55 kg/s)
Inlet Temperature	140°F (60°C)	72°F (22°C)
Outlet Temperature	104°F (40°C)	112°F (44°C)
Inlet Pressure	39 psi	39 psi
Inlet & outlet Pipe Diameter	7 inches	6 inches
Design Pressure	75 psig at 150°F	75 psig at 150°F
Total Heat Transfer Rate	5.5	5.5

4.2.2 Plate-and-Frame Heat Exchanger Description

The heat exchanger consists of 135 plates hanging vertically from the upper frame bar of the heat exchanger frame. Figure 4.2-2 and 4.2-3 show a side view of the heat exchanger and a front view of one of the plates, respectively. Primary system water flows into the heat exchanger from the inlet on the upper right of the face of the heat exchanger, flows down between every other plate, and exits at the lower right outlet. The secondary coolant flows in at the lower left inlet and flows up between every other plate, and exits at the upper left outlet. Figure 4.2-4 shows the flow paths for primary and secondary coolants. The gaskets are also shown in Figure 4.2-4. As can be seen, the chance of cross-talk is very small because there would have to be a leak in two different gaskets or a hole in one of the plates.

Plate-and-frame heat exchangers are modular in design. They are designed to be quickly and easily disassembled and cleaned. This means that if a gasket should develop a leak, the heat exchanger can be disassembled, the faulty gasket replaced, and the heat exchanger reassembled. Should a shell-and-tube heat exchanger develop a leak, the faulty tube has to be plugged, resulting in reduced capacity for the heat exchanger.

4.2.3 Cooling Towers

The cooling towers considered for this project are manufactured by the Marley Cooling Tower Company. Marley supplied the MITR-III design group with a computer code UPDATE, that takes the system requirements and weather conditions as input and returns cooling tower models that meet the requirements.

The main limitation in the choice of cooling tower is the space constraint. The present towers occupy a space of approximately 20 feet by 39 feet. Another important constraint is the ability to remove 12 MW with three or fewer unit cells. Of five series of

cooling towers offered by Marley, three series gave desirable operating conditions. However, series Sigma Wood and Sigma Steel models occupy at least 28 feet by 33 feet. Thus the NC series were selected for further analysis.

Another constraint is the maximum flow rate allowable through the secondary cooling system. This limit is set by the pipe size and also the design pressure of the heat exchangers. Higher flow rates will also require the use of bigger and more expensive cooling towers. Thus it is necessary to minimize the flow rate.

Another factor which also affects the design of the cooling system is the wet-bulb temperatures around the MITR. Data from Logan International Airport Weather Station has been analyzed. It was found that the wet-bulb temperature can be as high as 79°F (26°C). This narrowed the choice of cooling towers that can remove 12 MW with such a wet bulb temperature and still give desirable operating conditions. Mechanical design handbooks give a value of 75°F as the design wet-bulb temperature for summer air conditions in Boston. Actual value of this design parameter can normally be higher by 1 to 3°F, i.e. actual wet-bulb temperatures can be between 76°F to 78°F consistent with actual data from Logan Airport. Since the choice of design wet-bulb temperature largely affect the capability of the cooling tower in the summer, various wet bulb temperatures were considered in the selection.

Table 4.2-3
Selected Cooling Towers

Model	NC 7033	NC 8003	NC 8013	NC 8023
# of Cells	3	3	3	3
Overall Size				
Width x Length	21.9 x 35.9	21.9 x 32.9	21.9 x 32.9	21.9 x 32.9
Height, (ft)	13.3	19.8	19.8	19.8
Price Comparison	1.00	1.33	1.34	1.35
Energy Cost	1.75	1.00	1.15	1.44
Maximum flow rate (gpm)	2545	2598	2743	2973
Margin above wet-bulb temperature (°F)	4.3	4.1	3.8	3.5

All the above models have a width that is larger than the present basin by about 2 feet. Model NC 8003 is selected due to its efficient energy performance. It costs 33 % more than model NC 7033. Model NC 8003 is also selected because it is taller than model NC 7003. If better performance during hot summers are required, model NC 8013 or even NC 8023 should be considered. A price quote of \$104,100 was obtained for model NC 8003.

4.3 Results of Coupled Neutronic and Thermal Hydraulic Transient Analyses for a 10 MW MITR

A description of this study and its objective was presented in Section 3.6. The following is a summary of the results obtained from this study applicable to a 10 MW MITR.

4.3.1 MITR-III Model Assumptions

In modeling the MITR system for operation at 10 MW, the following assumptions were made:

- Two identical plate type heat exchangers are utilized under normal operating conditions [16].

- The pumps' head-capacity curve is given by the following polynomial:

$$\left(\frac{H}{H_r}\right)\left(\frac{\omega}{\omega_r}\right)^2 = 1.305 - 0.255 * X - 0.05 * X^2 \quad \text{where } X = \left(\frac{Q}{Q_r}\right)\left(\frac{\omega_r}{\omega}\right), \text{ and}$$

H = pump head (m)

ω = pump angular speed (rad/s)

Q = volumetric flow rate (gpm)

The subscript r represents the value at the pump's rated conditions.

- A secondary volumetric flow rate of 2290 gpm and a heat exchanger secondary inlet temperature between 25 and 27°C.
- The nuclear peaking factors for the assemblies under study were the following [17]:

	Assembly A-1	Assembly C-1
Radial Peaking Factor	1.388	1.131
Axial Peaking Factor	1.67 (at bottom) 1.05 (at top)	2.475 (at bottom)

- To account for fuel axial conductivity and entrance effects, the local (bottom) peaking factors were reduced by 15 percent.
- A hot-channel flow disparity of 0.916, an enthalpy-rise engineering hot-channel factor of 1.146, and a heat flux engineering hot-channel factor of 1.527 [17].
- A coolant temperature reactivity coefficient of $-0.01 \beta/^{\circ}\text{C}$, and a void reactivity feedback of $-0.6 \beta/^{\circ}\text{C}$.

4.3.2 Limiting Safety System Setting

First, the code TAMLOC (See Section 3.6 and Appendix C) was used to perform steady-state calculations to propose Limiting Safety System Settings (LSSS). The LSSS consists of the maximum outlet temperature, minimum flow rate, and maximum power such that if operations are constrained to within this envelope, subcooled nucleate boiling does not occur anywhere in the core.

Following the same approach as for MITR-II, the MITR-III power LSSS is taken to be 120 % of nominal power (12 MW). The flow LSSS is arbitrarily chosen to be 2500 gpm. Further analysis should be performed to determine the adequacy of this value. The core outlet temperature LSSS is set by determining the minimum temperature that would produce subcooled nucleate boiling within the core at 2500 gpm and 12 MW. This temperature is calculated to be 69°C. The following table summarizes the proposed Limiting Safety System Setting.

Table 4.3-1
Proposed Limiting Safety System Setting

Maximum Power (MW)	12
Minimum Flow Rate (gpm)	2500
Maximum Core Outlet Temperature (mixed-mean, °C)	69
Maximum Core Inlet Temperature (°C)	50

4.3.3 Operational Limits

In addition to the LSSS, operational limits were determined based on three transients: (1) a loss of heat sink, (2) a reactivity insertion, and (3) a loss of flow. The initial conditions were assumed to be the following:

- An initial power of 10 MW;
- A core inlet and outlet temperatures of 45.7°C and 59.7°C, respectively; and
- A primary volumetric flow rate of 2800 gpm.

The loss of heat sink was modeled as a rapid (less than 3 seconds) decrease in the secondary flow rate. From this calculation, the hot leg temperature operational limit was determined as the maximum hot leg temperature at which the reactor should scram to prevent the core outlet temperature from exceeding the LSSS (69°C). This hot leg temperature was calculated to be 62°C.

The reactivity insertion accident was postulated to determine the power operational limit. The accident postulated is a full shim bank withdrawal at maximum speed (4.25 in/min.). This is approximated as a ramp reactivity insertion of 12 dollars in 207 seconds. The power operational limit has been determined as the maximum power at which the

reactor should scram to prevent the power overshoot (due to the rod insertion time) from exceeding the power LSSS (12 MW). The power operational limit was calculated to be 11.5 MW based on a maximum control blade insertion time (from full out to 80 % insertion) of 0.8 seconds.

The loss of flow accident was postulated to determine the flow rate operational limit. In addition, restrictions on the maximum control blade insertion time and the maximum shim bank elevation have been determined from the results of this accident simulation. The flow rate operational limit has been determined as the minimum mass flow rate at which the reactor should scram as to prevent nucleate boiling prior to reactor shutdown. This accident scenario is the most limiting of the accidents studied thus far. A compromise between the maximum (initial) control shim bank height, maximum control blade insertion time, and minimum mass flow rate is required. The results of iterative calculations show that a maximum control blade insertion time (from full out to 80 % insertion) of 0.8 seconds, a maximum shim bank height of 19 inches (out of the core), and a minimum volumetric flow rate of 2650 gpm is a good compromise. The following table summarizes the proposed operational limits.

Table 4.3-2
Proposed Operational Limits

		Based on
Maximum Power (MW)	11.5	Overshoot from reactivity insertion not to exceed 12 MW (LSSS)
Minimum Flow Rate (gpm)	2650	No nucleate boiling prior to scram on a loss of flow accident
Maximum Hot Leg Temperature (°C)	62	Core outlet temperature not to exceed 69°C (LSSS) on a loss of heat sink accident
Maximum Control Blade Insertion time (full out to 80 % insertion, seconds)	0.8	No nucleate boiling prior to scram on a loss of flow accident
Maximum Shim Bank Height at Full-Power (inches above full insertion)	19	No nucleate boiling prior to scram on a loss of flow accident

It is important to note that (in the code validation section) for the MITR-II pump coastdown, the simulation under-predicted the mass flow rate in the early portion of the transient [6]. If data becomes available proving that a similar under-prediction occurs for the MITR-III pump coast-down simulation, the 0.8 maximum blade drop time and the 19 inches maximum shim bank height restrictions can be relaxed.

The following sections present the simulated behavior of the MITR-III system for several transients with the proposed trip settings of Table 4.3-2.

4.3.4 Loss of Heat Sink

The loss of heat sink simulation involves a complete loss of secondary pumping power. The flow in the secondary side of the heat exchangers was reduced from 2290 gpm to zero in a 30 second interval. The initial conditions were identical to the ones used in determining the operational limits (Section 4.3.3)

Due to the decrease in flow rate, the heat exchangers' energy removal rate drops significantly in a short period of time. As a consequence, the core inlet temperature and hot leg temperature rise sharply as shown in Figures 4.3-1 and 4.3-2, respectively. The reactor is scrammed upon reaching a 62°C hot leg temperature preventing the core outlet temperature, shown in Figure 4.3-3, from reaching 69°C (LSSS). For the time period simulated, the clad temperature never reaches the onset-of-nucleate-boiling temperature.

Figure 4.3-1. Core Inlet Temperature in Loss of Heat Sink Accident

Figure 4.3-2. Hot Leg Temperature in Loss of Heat Sink Accident

Figure 4.3-3. Core Outlet Temperature in Loss of Heat Sink Accident

4.3.5 Continuous Control Blade Withdrawal

The continuous single blade withdrawal simulation was simplified by assuming that the differential worth of the control blade is uniform and equal to the maximum differential worth for the most reactive control blade ($0.147 \beta/\text{inch}$). A total control blade reactivity worth of two dollars was used [18]. Two blade withdrawal velocities were studied: 4.25 and 8.50 inches/min. This is equivalent to inserting two dollars of reactivity in ramps of 192 and 96 seconds, respectively. The initial conditions were identical to the ones used in determining the operational limits (Section 4.3.3).

Note that, for both reactivity ramps, the reactor scrams upon reaching 11.5 MW and the overshoot never reaches the Limiting Safety System Setting of 12 MW (Figures 4.3-4 and 4.3-5). The core inlet and hot leg temperatures remain approximately constant throughout the transients until the reactor scrams. The core outlet temperature is shown in Figures 4.3-6 and 4.3-7. Figures 4.3-8 and 4.3-9 show that the clad temperature stays well below the onset-of-nucleate-boiling temperature (approximately 117°C) for both cases.

Figure 4.3-4. Core Power Generation in Blade Withdrawal Accident (4.25 inches/min.)

Figure 4.3-5. Core Power Generation in Blade Withdrawal Accident (8.50 inches/min.)

Figure 4.3-6. Core Outlet Temperature in Blade Withdrawal Accident (4.25 in./min.)

Figure 4.3-7. Core Outlet Temperature in Blade Withdrawal Accident (8.50 in./min.)

Figure 4.3-8. Clad Temperature (hot-spot) in Blade Withdrawal Accident (4.25 in./min.)

Figure 4.3-9. Clad Temperature (hot-spot) in Blade Withdrawal Accident (8.50 in./min.)

4.3.6 Pump Coastdown

The pump coastdown is simulated as a complete loss of electrical power to the pump motor. The coolant flow does not seize instantaneously since the pump rotor/impeller and the fluid within the loop have momentum. Rather, the coolant flow rate "decays" to zero in approximately 21 seconds as shown in Figure 4.3-10. When the coolant flow rate reaches 2650 gpm, the reactor is scrammed (Figure 4.3-11).

During a pump coastdown, both fluid and clad experience two temperature peaks. The first temperature peak is due to the rapidly decreasing flow while the control blades are not yet inserted in the core. The results of this simulation indicate that, for the first peak, the most limiting clad temperature occurs at the bottom of assembly C-1. The behavior of the clad temperature at this spot is shown in Figure 4.3-12. The second peak occurs after the reactor scrams and it is due to the coolant flow rate reaching a minimum value while in transition between forced flow and natural circulation while the decay heat is still relatively large. The most limiting clad temperature for this case occurs at the top of assembly A-1. Nucleate boiling is predicted to occur at this spot 26.5 seconds after the pumps begin to coast down. The calculations, however, are followed just a few seconds past the point where nucleate boiling begins since, due to possible flow instabilities (a phenomenon not included in this model), the results may not be accurate beyond the onset of nucleate boiling. Therefore, due to the short simulation time, the second clad temperature peak of assembly A-1 is only approached in Figure 4.3-13. The core outlet temperature is shown in Figure 4.3-14.

The natural circulation valves are predicted to open approximately 11 seconds after the pumps begin to coast down. In the short time of simulation, the natural circulation flow does not reach its maximum value as shown in Figure 4.3-15.

Figure 4.3-10. Loop Coolant Mass Flow Rate in Pump Coastdown Accident

Figure 4.3-11. Core Power Generation in Pump Coastdown Accident

Figure 4.3-12. Clad Temperature (hot-spot, assembly C-1) in Pump Coastdown Accident

Figure 4.3-13. Clad Temperature (hot-spot, assembly A-1) in Pump Coastdown Accident

Figure 4.3-14. Core Outlet Temperature in Pump Coastdown Accident

Figure 4.3-15. Natural Circulation Coolant Flow Rate in Pump Coastdown

4.3.7 Summary

Table 4.3-1 shows the Limiting Safety System Setting proposed. If steady-state operations are constrained to within this envelope, subcooled nucleate boiling is not expected to occur anywhere in the core. In addition, operational limits were determined based on the reactor behavior to a loss of flow, a reactivity insertion, and a loss of heat sink transient. The operational limits proposed are summarized in Table 4.3-2.

The simulated behavior of the MITR system to a loss of flow, a reactivity insertion, and a loss of heat sink transient show that satisfactory operation can be achieved if operations are maintained within the LSSS and operational limits. An ongoing study aimed at analyzing transients with a detailed core model will be used to update the limits proposed in this study (see Appendix D).

4.4 Cost Analysis of Upgrade to 10 MW

The methodology of this study was presented in Section 3.7. The following is a summary of the results obtained from this study applicable to upgrading the MITR to 10 MW.

4.4.1 Engineering and Design Costs

Engineering and Design costs are summarized in Table 4.4-1. Engineering and Design costs are found by multiplying the number of weeks required to complete a project by the weekly salary of an MIT Research Assistant (\$300). Factors for employee benefits and indirect expenses are also included. Additionally, it is assumed that a faculty member will spend 10% of his time supervising each graduate student's work. This requires that 10% of a faculty salary be included in the cost as well (however, overhead on faculty salaries was not included in this cost analysis). Proposed changes in the structure of student assistantships may require that the student's tuition be covered by the project in the future. In this report student costs have been calculated using the current 1994 approach.

Table 4.4-1
Engineering and Design Costs For MITR-III At 10 MW

ENGINEERING AND DESIGN CATEGORY	COST (1994 DOLLARS)
Thermal Hydraulics	\$180,000
Reactor Physics/Fuel Management	\$180,000
Shielding, Dose, Waste	\$45,000
Fuel Management	\$60,000
Materials Integrity	\$30,000
Documents	\$15,000
TOTAL	\$510,000

*The total with overheads included would be approximately \$565,000

Because the reactor will be relicensed at a greater power level there is a considerable amount of design work that must be done to verify that the system can safely operate at 10 MW. Thermal-hydraulic and neutronic analysis methods are being developed that are applicable to the general MITR design at either 5 or 10 MW [6,19]. Once a decision is made to pursue one of these two options, the engineering and design work can be focused on the components needed to meet that power level.

These tools are currently being used to estimate optimum design parameters for heat exchangers, pumps, and cooling towers for a 10 MW reactor. The Engineering and Design work described here will also be used to update the SAR and Technical Specifications of MITR-III, and to evaluate the impact of the new components on system safety.

Some of the thermal hydraulic calculations that were done for the original licensing will be repeated using newer methods of analysis. In particular, the Limiting Safety System Settings and Operating Limits will be reevaluated using transient models of the MITR-III primary and secondary systems. New limits will be derived for 10 MW operation based on criteria similar to those used for MITR-II at 10 MW. These analyses will look in detail at the onset of nucleate boiling, the stability of the flow through the core (Ledinegg Instability), and the reaction of the system to transients [6,19].

The reactor physics calculations done for MITR-II will be repeated. The same fuel element design will be used, but the higher power level will require the use of a high burnup fuel and/or a new system of fuel management to lengthen the time between refuelings. This work requires detailed knowledge of fuel cross sections and other physical parameters. Research is being done to study the advantages of new fuel types such as Silicide Dispersion Fuel and Extended Life Aluminide Fuel [ELAF] [17].

The MITR-II system of biological shielding and dose calculation will be reevaluated to ensure that it is adequate for 20 years of safe operation at 10 MW. The higher flux will mean that the dose rate through unprotected channels would roughly double. Radiation surveys and calculations are being done to determine if the current biological shield will be adequate to keep dose levels within limits at 10 MW. In addition, the increased power will lead to a larger amount of heat being deposited in the lead shield. Research will be done to determine if the current shield coolant system is adequate to remove this heat.

Some more studies will be done to verify the material integrity of the reactor systems. This includes aluminum aging studies on the reactor core and reflector tanks, and an analysis of the primary piping system.

The results of the above research will be used to make update documents such as the Standard Operating Procedure/Abnormal Operating Procedure (SOP/AOP), Operator Requalification Program, Safety Analysis Report (SAR), and Technical Specifications. The development of a thorough Quality Assurance (QA) program to oversee the replacement of components is also included under documents.

4.4.2 Operating Costs

The Operating Costs associated with upgrading and licensing the reactor are shown in Table 4.4-2. The decision to use in-house versus subcontracted labor is based on a trade-off between labor costs and reactor down-time. It is assumed for this analysis that it is most economical to use the NRL staff that is currently on the payroll for most of the work because they are familiar with the systems. Hiring outside contractors to install components and perform some maintenance may reduce the overall down-time slightly, but it is unlikely that this savings will be sufficient to make this option economical.

Table 4.4-2
Operating Costs For MITR-III At 10 MW

OPERATING COST CATEGORY	COST (1994 DOLLARS)
Technical Specialists	\$194,800
Support Equipment	\$15,000
Lost Income	\$667,000
Payroll and Operation	\$601,300
TOTAL	\$1,478,100

There may, however, be a need to hire outside draftsmen, engineers, consultants, or technicians to perform certain tasks. One technician, one electrical consultant, and one draftsman (who will also oversee the QA program) are included in this estimate. Also included is the cost of hiring an independent auditor to aid in the licensing process.

Support equipment includes purchases that must be made to aid in the completion of Direct Cost and Engineering and Design Costs. This may include computers, software, machining tools, or testing and calibration equipment.

The lost income included in this estimate is based on a four month total down-time. This time could be lengthened by shipping delays from manufacturers, components that do not fit, or licensing delays. Because the replacement of primary loop components can be done without a long shutdown, the schedule is set by the time needed to upgrade the secondary system (piping and cooling towers). A logical work schedule and an effective QA program will be implemented to reduce the reactor down-time. For example, the reactor should continue to run until the replacement components have all arrived and passed an in-house QA inspection.

The payroll of the reactor laboratory must be paid while any refurbishments are being done. The Payroll and Operation category includes the salaries of NRL employees working on the upgrade, permits, services, and insurance. The estimated operating cost chargeable to the upgrade is based on a four month time period.

The costs of site preparation and cleanup are included in the cost of the major components found in Table 4.4-3.

4.4.3 Other Direct Costs

In order to license the MITR at 10 MW and to allow it to operate for twenty years, many key components will need to be replaced. Preliminary design work done over the past two years has provided the technical specifications necessary to make rough estimates of the cost. The estimated direct costs are shown in Table 4.4-3.

Table 4.4-3
Other Direct Costs For MITR-III At 10 MW

DIRECT COST CATEGORY	COST (1994 DOLLARS)
Primary Heat Exchangers	\$48,900
Primary Pumps	\$104,100
Secondary Pumps	\$104,100
Secondary Piping	\$90,100
Cooling Towers & Filtration System	\$256,000
Control System	\$47,500
Containment	\$139,100
Effluent Release Stack	\$5,000
Electrical & Instruments	\$280,000
ECCS System	\$10,000
Shield Cooling	\$50,000
Decontamination/Waste Disposal*	\$95,000
TOTAL	\$1,229,800

* Complete waste disposal costs cannot be estimated until a low-level waste site can be designated.

The three main heat exchangers currently in place in MITR-II will need to be replaced with larger units to handle the doubled heat load. This updated analysis uses cost estimates obtained from Alpha-Laval plate and frame heat exchangers. The Alpha-Laval models take up less space and cost much less than shell and tube models (like those currently used in MITR-II). A study by M. Arno suggests that plate and frame heat exchangers are relatively easy to disassemble for routine maintenance, and that there is no cross-leakage problem between the primary and secondary sides [20]. Operational experience at the National Bureau of Standards and Technology (NIST) has shown that any leaks that may develop would be external rather than between fluid systems. Three heat exchangers, each with 50% (i.e., 5.5 MW) capacity, will be installed. NIST has reported some difficulty with fouling on the secondary side, so a new secondary filtration system may need to be installed.

The primary and secondary system pumps (two of each) will need to be replaced with larger units. Preliminary analyses have been performed to determine the best combination of primary and secondary flow rates to use. These flow rates, along with heat exchanger and cooling tower performance, will define the key parameters for the new heat removal system. Based on the required pump heads and flow rates, Byron Jackson Inc., has recommended a set of primary pumps that is being used to model the 10 MW system [6,19]. The possible use of magnetic bearings is being studied to see if the improved performance and ease of maintenance would be advantageous to the project.

The underground secondary piping and electrical lines from the containment building to the cooling towers need replacement. These pipes have shown some leaks and are inaccessible in their current configuration. The current pipes to and from the cooling

towers will be removed and replaced. The same channel will be used to run a conduit with new electrical wires for power and control to the towers. It would be desirable to cover this channel with a removable lid to allow access for maintenance and repair work to be carried out without having to dig the pipes up again. This would require a set of pipe heaters to be available in the winter.

The two wooden cooling towers that are currently in use were installed 35 years ago. They are inadequate to remove the estimated 11 - 12 MW of heat that would be sent to them from the primary and auxiliary systems. Marley Cooling Tower Co. has provided the MIT Nuclear Reactor Laboratory (NRL) with their recommendation for a system that would handle the heat load from a 10 MW reactor. This recommendation is based on the restricted physical envelope (due to the current tower basin size), operating temperature range, and secondary flow rate provided by the NRL. The Marley system is a set of three modular, stainless-steel heat exchangers and associated fans and piping [20]. Also included in the cooling tower estimate is the cost of installing a filtration system and a fire alarm system.

The control blade drive motors that are used to raise the six control blades and one regulator rod are old and have no commercially available spare parts. These motors need to be replaced with similar models. The control system itself will be upgraded.

The containment building is pressure-tested periodically for leaks and spot-checked for corrosion near the ground. A recently renewed anode system is being used to protect against future corrosion. The truck lock will require considerable work to allow it to be usable. Two gaskets, a seating surface, and structural portions of the lock need to be repaired. This cost estimate assumes that the containment will be inspected and any corrosion damage will be repaired, and that the truck lock will be made serviceable.

The electrical system components, including emergency batteries, switch boxes, wiring, transformer, and generator, need refurbishment or replacement. Only rough estimates are available for the large components because it is unclear what the electrical requirements will be for the reactor system with its new components. The motor control centers are not included in this cost analysis.

The ECCS system that was designed for MITR-II will be upgraded to reflect the new power level. A study by Parra indicates that a larger pump or another nozzle will need to be installed in the system [17].

With an extra 5 MW of heat being generated, a study will be done to determine whether or not the current shield coolant loop will need to be expanded. Additionally, Parra's analysis shows that a larger pump will need to be installed to deliver the required flow rate for 10 MW operation.

Decontamination and waste disposal costs for all components that need to be replaced are included together. Secondary side components (pumps, piping, etc.) can be disposed of by conventional means. The cooling towers will be dismantled and disposed

of. The heat exchangers and primary side components will need to be evaluated for decontamination prior to disposal. Until a low level waste depository will accept these shipments, it is not possible to assign a disposal cost to these components. There is adequate on-site storage for these heat exchangers.

4.4.4 Summary of Costs To Upgrade To 10 MW

Based on the assumption that a relicensed reactor will be operated for at least 20 years, the necessary design, operating, and direct component costs have been estimated as described above. As previously noted, these costs are sensitive to several variables:

- 1) The component replacement schedule will determine the required reactor down time. Delays from manufacturers and installation problems can be minimized with a thorough quality assurance program. Licensing and start-up delays are more difficult to control. However, modifications would proceed only after NRC authorization; i.e. a new license is granted.
- 2) The prices for major components will change as system parameters are modified during the ongoing design process. These costs are based on estimates from only a small number of sources. Once a detailed final design is settled on and the licensing requirements are firmly set it will be possible to obtain competitive bids on these components.
- 3) Before or during the time the upgrade takes place other systems or components could be found to be in need of replacement. In particular, the shield coolant and coolant purification systems will be analyzed to see if they need to be upgraded to handle the increase reactor power and primary flow rate.

The current analysis includes "best estimate" costs for all work outlined above. A 25% global contingency factor has been included to account for unforeseen work or price overruns. The following systems have been assumed to be adequate for future operation, and have not been included in this analysis:

- 1) Coolant Purification System and Ion Columns
- 2) Core and Reflector Tanks
- 3) Primary Piping System
- 4) Spent Fuel Pool
- 5) Graphite Reflector
- 6) Reflector Coolant System
- 7) Containment Building and Reactor Shielding

Table 4.4-4 shows a summary of the estimated costs for upgrading the reactor to 10 MW.

Table 4.4-4
Summary of Costs For Upgrading to 10 MW

COST CATEGORY	COST (1994 DOLLARS)
Engineering and Design	\$510,000
Operating Costs	\$1,478,100
Other Direct Costs	\$1,229,800
Contingency	\$804,500
TOTAL	\$4,022,400

5 SUMMARIES OF RESULTS APPLICABLE TO RELICENSE A 5 MW MITR

5.1 Steady-State MITR-II Operation at 5 MW

The MITR-II has been operable at power levels up to 5 MW since 1976. The steady-state operating conditions as established by the Technical Specifications have been found to be satisfactory. The Limiting Safety System Setting have been set to avoid nucleate boiling in the core and are given as shown in the following table.

Table 5.1-1
MITR-II Limiting Safety System Setting

	2 Primary Pumps	1 Primary Pump	0 Primary Pump
Minimum Flow Rate (gpm)	1800	900	0
Maximum Power (MW)	6.0	3.0	0.10
Minimum Height of Water Above the Core	4'' below overflow pipe	4'' below overflow pipe	4'' below overflow pipe
Maximum Outlet Temperature (°C)	60	60	60

In the following sections, a reassessment of these limits and a general review of the transient response is given. In addition, an upgraded secondary-system for 5 MW operation has been designed and the specifications are given in the following section.

5.2 Results of Secondary System Design for 5 MW Operation

5.2.1 Secondary-Side Design Specifications

Even if MITR operations are continued at 5 MW, the heat exchangers and the cooling towers should be replaced. In this study, the replacement options for the heat exchangers and the cooling towers were investigated in detail. Further study is necessary to determine the secondary pumping capacity. Table 5.2-1 lists the secondary system design requirements to operate at 5 MW.

Table 5.2-1
Secondary System Specifications

Flow rate (gpm)	1250
Total heat load on cooling towers (MW)	6.0
Max. heat exchanger load, primary-to-secondary (MW)	5.5
Max. auxiliary load (AC, D ₂ O, Shield HX's) (MW)	0.5
D ₂ O heat exchanger load (MW)	0.25
Shield heat exchanger load (MW)	0.25
Cold leg temperature (°C)	27
Hot leg temperature (°C)	45.4

The schematic of the secondary system flow path is shown in Section 4.2 (Figure 4.2-1).

5.2.2 Cooling Towers

The current wooden cooling towers are more than 35 years old. Severe weather exposure has caused deterioration in the structural integrity of the towers and also its heat removal capability. The cooling towers must be replaced for continued operation beyond 1996.

The cooling towers considered for this project are manufactured by the Marley Cooling Tower Company. Marley supplied the MITR-III design group with a computer code UPDATE, that requires tower flow rate, hot and cold water temperatures and wet bulb temperatures as input. UPDATE returns several cooling tower models that meet the requirements provided by the user.

The main limitation in the choice of cooling tower is the space constraint. The present towers occupy a space of approximately 20 feet by 39 feet. Of five series of cooling towers offered by Marley, three series gave desirable operating conditions.

However, series Sigma Wood and Sigma Steel models occupy at least 28 feet by 33 feet. Thus, the NC series were selected for further analysis.

Another important consideration in selecting the cooling tower model is the fact that this MITR-III project may initially decide to relicense a 5-MW reactor and at a later date upgrade to a 10-MW reactor. This means that for the 5-MW reactor, a 2-unit cell must be selected such that a third similar cell can be added at a later date to provide for the 10-MW reactor. The selections below are based on calculations for a 3-cell cooling towers for the 10-MW reactor.

Table 5.2-2
Selected Cooling Towers

Model	NC 7032	NC 8002	NC 8012	NC 8022
# of Cells	2	2	2	2
Overall Size				
Width x Length (ft)	21.9 x 23.9	21.9 x 21.9	21.9 x 21.9	21.9 x 21.9
Height (ft)	13.3	19.8	19.8	19.8
Price Comparison	1.00	1.33	1.34	1.35
Energy Cost	1.75	1.00	1.15	1.44
Maximum flow rate (gpm)	1697	1732	1829	1982
Margin above wet-bulb temperature (°F)	4.3	4.1	3.8	3.5

From UPDATE, the results indicate that Model NC 8012 is preferable to Model NC 7032 and Model NC 8022. The required flow rate is only 1250 gpm for the 5-MW reactor. Models NC 8012 and NC 8022 are also capable of running at higher wet-bulb temperatures than the other models as can be seen from their lower temperature margin above the wet-bulb temperature. This means that the second and the third model can maintain a cold water temperature of 80.0 °F at wet-bulb temperatures of about 76.5°F where as model NC 7032 can only operate at wet-bulb temperatures of about 75.5°F.

5.2.3 Heat Exchangers

The three primary heat exchangers are barely adequate for 5 MW and during hot, humid weather, reactor power must be reduced. Also, a number of tubes have been plugged in one of the existing units. This heat exchangers must be replaced for continued operation.

The following procedure will be used to estimate the heat exchanger temperature differentials for a plate-and-frame type heat exchanger. This type of heat exchanger was described in Section 4.2.

- a) Calculate flow rates and velocities across the plates.

- b) Determine the Reynolds, Prandtl and Nusselt numbers for the flows.
- c) Using the average heat flux transferred, determine the temperature difference across the fluid to plate interface.
- d) Using the resistance across the plates, determine the plate temperature differential.
- e) Sum to determine an average heat exchanger temperature differential.

The following data for the plate heat exchangers and water at 112°F (50°C) were used:

	Primary Side (~120°F or 49°C)	Secondary Side (~98°F or 37°C)
Flow rate	1250 gpm (77.9 kg/s)	1250 gpm (78.3 kg/s)
Heat exchanged	6.0 MW (20.5 MBTU/hr)	6.0 MW (20.5 MBTU/hr)
Inlet temperature	134.5°F (56.9°C)	80.0°F (26.7°C)
Outlet temperature	104.0°F (40.0°C)	110.4°F (43.6°C)

The plate heat exchanger design specifications used to calculate the log mean temperature difference are as follows:

Volume of plate heat exchanger primary and secondary sides	44.2 gallons	0.167 m ³
Diameter of plate heat exchanger plena	6 inches	15.24 cm
Plate width	24 inches	0.61 m
Plate height	50.8 inches	1.29 m
Plate heat exchanger thickness (estimated)	24 inches	0.61 m

The above parameters give a value of 27°F (15.0°C) for the ΔT_{lm} (Log Mean Temperature Difference).

A quotation from Alfa-Laval has been obtained by Arno [20]. The heat exchanger is quoted at \$14,000 each.

5.3 Results of Coupled Neutronic and Thermal Hydraulic Transient Analyses for a 5 MW MITR

The methodology of this study was presented in Section 3.6. The following is a summary of the results obtained from this study applicable to a 5 MW MITR.

5.3.1 Model Assumptions

In modeling the MITR system for operation at 5 MW, the following assumptions were made:

- Two identical shell-and-tube heat exchangers are utilized under normal operating conditions.
- The pumps head-capacity curve is given by the following polynomial:

$$\left(\frac{H}{H_r}\right)\left(\frac{\omega_r}{\omega}\right)^2 = -2.56 + 14.25 * X - 15.31 * X^2 + 4.6 * X^3 \quad \text{where } X = \left(\frac{Q}{Q_r}\right)\left(\frac{\omega_r}{\omega}\right)$$

H = pump head (m)

ω = pump angular speed (rad/s)

Q = volumetric flow rate (gpm)

The subscript r represents the value at the pump's rated conditions

- A secondary volumetric flow rate of 1660 gpm and a heat exchanger secondary inlet temperature of 26.6°C.
- The nuclear peaking factors for the assemblies under study were the following [17]:

	Assembly A-1	Assembly C-1
Radial Peaking Factor	1.388	1.131
Axial Peaking Factor	1.67 (at bottom) 1.05 (at top)	2.475 (at bottom)

- To account for fuel axial conductivity and entrance effects, the local (bottom) peaking factors were reduced by 15 percent.
- A hot-channel flow disparity of 0.916, an enthalpy-rise engineering hot-channel factor of 1.146, and a heat flux engineering hot-channel factor of 1.527 [17].
- A coolant temperature reactivity coefficient of $-0.01 \beta/^{\circ}\text{C}$, and a void reactivity feedback of $-0.6 \beta/^{\circ}\text{C}$.

The results obtained for several transients simulated using TAMLOC are presented in the following sections. A core power of 5 MW and a volumetric flow rate of 2050 gpm were assumed as the initial conditions for every case studied. The simulations were performed with reactor trips set according to the present Limiting Safety System Setting (LSSS): 60°C hot leg temperature, 1800 gpm flow rate, 10 second reactor period, 6 MW power, and 2.9 meter water level. In the MITR-II, the actual trip settings are set conservative relative to the LSSS (i.e., 55°C, 1960 gpm, 5.5 MW, 11 sec. period). Although the present design includes a secondary flow rate trip, it was not included in the model.

5.3.2 Loss of Heat Sink

The loss of heat sink accident studied involved a complete loss of secondary pumping power. To simulate this accident, the secondary coolant flow was reduced from the rated volumetric flow rate to zero flow rate in a five second interval.

Due to the decrease in secondary flow rate, the heat exchangers' energy removal rate drops significantly in a short period of time as shown in Figure 5.3-1. As a consequence, the core inlet temperature rises from approximately 46°C to 58°C in about a minute as shown in Figure 5.3-2.

Figure 5.3-1. Heat Exchangers Energy Removal Rate in Loss of Heat Sink Accident

Figure 5.3-2. Core Inlet Temperature in Loss of Heat Sink Accident

This sharp increase in core inlet temperature produces a negative reactivity feedback decreasing the power from 5.0 MW to 3.6 MW in approximately 50 seconds as shown in Figure 5.3-3. However, since the energy removal rate decreases faster than the feedback effects, the core outlet temperature rises sharply from approximately 56°C to 63°C in about 50 seconds. During this time, the water leaving the mixing pool above the core into the hot leg piping rises from approximately 55°C to 60°C reaching the hot leg temperature trip setting and, consequently, scrambling the reactor 50 seconds after the transient was initiated. Therefore, the core outlet and hot leg temperatures momentarily drop sharply. However, due to the decay heat generated in the core both temperatures slowly increase thereafter. Consequently, the core inlet temperature slowly increases as well. The core outlet and hot leg temperatures are shown in Figures 5.3-4 and 5.3-5, respectively. The clad temperature stays well below the value necessary for the onset of nucleate boiling.

Figure 5.3-3. Core Power Generation in Loss of Heat Sink Accident

Figure 5.3-4. Core Outlet Temperature in Loss of Heat Sink Accident

Figure 5.3-5. Hot Leg Temperature in Loss of Heat Sink Accident

It is important to note that the heavy-water tank surrounding the core tank has not been modeled here. This would contribute to the decay heat removal by conduction through the core tank. In addition, the secondary system was designed so that natural circulation can exist in the case of a loss of secondary pumping power. This occurrence was also excluded from the modeling. As a consequence, the long-term results are believed to be conservative.

5.3.3 Continuous Control Blade Withdrawal

The continuous single blade withdrawal simulation was simplified by assuming that the differential worth of the control blade is uniform and equal to the maximum differential worth for the most reactive control blade ($0.147 \beta/\text{inch}$). The maximum control blade velocity is 4.25 inches/min. and a total control blade reactivity worth of two dollars was used [18]. This is equivalent to inserting two dollars of reactivity in a ramp of 192 seconds.

Figure 5.3-6 shows the trend of the core power generation. Note that the reactivity insertion begins after 10 seconds of steady-state operation. The power rises from approximately 5.0 MW to 6.0 MW in about 10 seconds where it reaches the overpower trip setting and consequently scrams. The core outlet temperature experiences a sharp increase that is abruptly interrupted by the decrease in power as shown in Figure 5.3-7. The hot leg temperature (Figure 5.3-8) experiences a slight increase and then it decreases sharply following the reactor scram.

Figure 5.3-6. Core Power Generation in Blade Withdrawal Accident (4.25 inches/min.)

Figure 5.3-7. Core Outlet Temperature in Blade Withdrawal Accident (4.25 inches/min.)

Figure 5.3-8. Hot Leg Temperature in Blade Withdrawal Accident (4.25 inches/min.)

The clad temperature (hot-spot) increases from 81°C to 88°C approximately in 10 seconds. Once the reactor scrams, the clad temperature drops as shown in Figure 5.3-9. The clad temperature (hot-spot) stays well below the onset-of-nucleate-boiling temperature.

Figure 5.3-9. Clad Temperature (hot-spot) in Blade Withdrawal Accident (4.25 in./min.)

A control blade withdrawal velocity of twice the maximum velocity (8.50 in./min.) was also studied. This is equivalent to inserting two dollars of reactivity in a ramp of 96 seconds.

In this case the power increases from 5 MW to 6 MW in approximately 5.5 seconds as shown in the following figure. The core outlet, hot leg inlet, and clad (hot-spot) temperatures are shown in Figures 5.3-11, 5.3-12, and 5.3-13, respectively. The clad temperature never reaches the onset-of-nucleate-boiling temperature.

Figure 5.3-10. Core Power Generation in Blade Withdrawal Accident (8.50 inches/min.)

Figure 5.3-11. Core Outlet Temperature in Blade Withdrawal Accident (8.50 inches/min.)

Figure 5.3-12. Hot Leg Temperature in Blade Withdrawal Accident (8.50 inches/min.)

Figure 5.3-13. Clad Temperature (hot-spot) in Blade Withdrawal Accident (8.50 in./min.)

5.3.4 Pump Coastdown

This section includes results of a pump coastdown simulation using TAMLOC as well as a comparison with available data: (1) a series of pump trip temperature measurements made during the MITR-II step-wise startup to full power [21], and (2) recent experimental flow measurement during a pump coastdown [22].

The comparison between the simulation results and the available data is shown in Figures 5.3-14, 5.3-15, and 5.3-16. The simulation results are in good agreement with the experimental values except for the core inlet temperature. This discrepancy may be due to the definition of the core inlet. In the simulation, the core inlet temperature is taken as the temperature of the fluid after going through the downcomer and mixing with the natural circulation flow. In a pump coastdown transient, this temperature will behave significantly different from the cold leg exit temperature.

Figure 5.3-14. Coolant Mass Flow Rate in Pump Coastdown Accident

Figure 5.3-15. Core Outlet Temperature in Pump Coastdown Accident

Figure 5.3-16. Core Inlet Temperature in Pump Coastdown Transient

Once the loop mass flow rate reaches about six percent of rated flow, the natural circulation valves are expected to open. At that instant the static pressure below the natural circulation valve exceeds that of the top, creating an upward flow through the valves bypassing the core. However, the rapidly decreasing pressure below the natural circulation valves produces a calculated reversal of flow through the valves 10 seconds later. Thereafter, the calculated natural circulation increases and reaches a maximum of approximately 1.8 kg/s at about 150 seconds from the initiation of the pump coastdown. Then, the natural circulation flow rate decreases as the heat slowly decays away. The natural circulation flow rate is shown in the following figure.

Figure 5.3-17. Natural Circulation Flow Rate in Pump Coastdown Accident

Immediately after the pumps begin to coast down, the core outlet temperature increases sharply from approximately 51°C to 53°C due to the decrease in core flow rate (this is not shown in Figure 5.3-15 due to low printing resolution). However, the 1800 gpm trip setting is reached very quickly interrupting this temperature increase. For a short period of time (about 3 seconds), the sudden decrease in power due to the reactor scram outweighs the decrease in flow rate, thus lowering the core outlet temperature. Eventually, the core flow rate reaches a minimum and the core outlet temperature begins to increase once again. In addition, the core inlet temperature increases since the source of coolant for natural circulation is the mixing pool above the core which is hotter than the coolant previously received from the cold leg. As a consequence of these two combined effects, the core outlet temperature increases at a rapid rate. Since the natural circulation flow rate increases and decay heat decreases, the core outlet temperature reaches a maximum of approximately 81°C at about 37 seconds from the pump coastdown initiation. The calculated clad temperature follows a similar trend as the core outlet temperature as shown in Figure 5.3-18. The clad temperature never reaches the onset-of-nucleate-boiling temperature.

Figure 5.3-18. Clad Temperature (hot-spot) in Pump Coastdown Accident

5.3.5 Summary

The transient response of the MITR operated at 5 MW has been studied using the TAMLOC code. The transients investigated included loss of heat sink, continuous control blade withdrawal, and pump coastdown.

An assessment of the current Limiting Safety System Setting (LSSS) was done by using the LSSS as the trip setting for all of the transients that were investigated. It was shown that the onset of nucleate boiling does not occur in any of the transients studied.

5.4 Cost Analysis of Relicensing At 5 MW

The methodology of this study was presented in Section 3.7. The following is a summary of the results obtained from this study applicable to relicensing the MITR at 5 MW.

5.4.1 Engineering and Design Costs

Engineering and Design costs are summarized in Table 5.4-1. Most of the design work will be done by MIT students, professors, and NRL engineers and staff. Engineering and Design costs are found by multiplying the number of weeks required to complete a project by the weekly salary of an MIT Research Assistant (\$300). Factors for employee benefits (1.435) and indirect expenses (1.620) are also included. Additionally, it is assumed that a faculty member will spend 10% of his time supervising each graduate student's work. This requires that 10% of a faculty salary be included in the price as well (however, there is no overhead on faculty salaries). Proposed changes in the structure of student assistantships may require that the student's tuition be covered by the project in the future. In this report student costs have been calculated using the current 1994 approach.

Table 5.4-1
Engineering and Design Costs For Relicensing MITR-II

ENGINEERING AND DESIGN CATEGORY	COST (1994 DOLLARS)
Thermal Hydraulics	\$120,000
Reactor Physics/Fuel Management	\$120,000
Shielding, Dose, Waste	\$30,000
Materials Integrity	\$30,000
Documents	\$6,000
TOTAL*	\$306,000

*The total with overheads included would be approximately \$336,600

Although the reactor will be relicensed at its current power level there is a considerable amount of design work that must be done. Thermal-hydraulic and neutronic analysis methods are being developed that are applicable to the general MITR design at either 5 or 10 MW [6,19]. Once a decision is made to pursue one of these two options, the engineering and design work can be focused on the components needed to meet that power level.

These tools are currently being used to estimate optimum design parameters for heat exchangers, pumps, and cooling towers for a 5 MW reactor. The Engineering and Design work described here will also be used to verify and/or update the SAR and Technical specifications of MITR-II, and to evaluate the impact of the replacement components on system safety.

Some of the thermal hydraulic calculations that were done for the original licensing will be repeated using newer methods of analysis. In particular, the Limiting Safety System Settings and Operating Limits will be reevaluated using transient models of the MITR-II primary and secondary systems.

The reactor physics calculations will be similarly reviewed. The same fuel element design will be used, but research is being done to study the advantages of other fuel types which are becoming available and can presumably be used in the unmodified core of the MITR-II to provide increased safety or efficiency (e.g., Silicide Dispersion Fuel, Extended Life Aluminide Fuel [ELAF]) [17]. Work is also being done to optimize the fuel management program that is used in MITR-II and to find better ways to calculate core physical parameters.

The MITR-II system of biological shielding and dose calculation will be reevaluated to ensure that it is adequate for 20 more years of safe operation at 5 MW. This work will include experimental validation of the existing biological shield and studies of the shield coolant system. No major redesign work is anticipated in this area.

Some studies will be done to verify the material integrity of the reactor systems. This includes aluminum aging studies on the reactor core and reflector tanks, and an analysis of the primary piping system.

The results of the above research will be used to make any necessary revisions to documents such as the Standard Operating Procedure/Abnormal Operating Procedure (SOP/AOP), Operator Requalification Program, Safety Analysis Report (SAR), and Technical Specifications. The development of a thorough Quality Assurance (QA) program to oversee the replacement of components is also included under documents.

5.4.2 Operating Costs

The Operating Costs associated with relicensing the current reactor are shown in Table 5.4-2. The decision to use in-house versus subcontracted labor is based on a trade-off between labor costs and reactor down-time. It is assumed for this analysis that it is most economical to use the NRL staff that is currently on the payroll for most of the work because they are familiar with the systems. Hiring outside contractors to install components and perform some maintenance may reduce the overall down-time slightly, but it is unlikely that this savings will be sufficient to make this option economical.

Table 5.4-2
Operating Costs For Relicensing MITR-II

OPERATING COST CATEGORY	COST (1994 DOLLARS)
Technical Specialists	\$192,000
Support Equipment	\$15,000
Lost Income	\$500,000
Payroll and Operation Costs	\$451,000
TOTAL	\$1,158,000

There may, however, be a need to hire outside draftsmen, consultants, engineers, or technicians to perform certain tasks. One technician and one draftsman (who will also oversee the QA program) are included in this estimate. Also included is the cost of hiring an independent auditor to aid in the licensing process.

Support equipment includes purchases that must be made to aid in the completion of Direct Cost and Engineering and Design Costs. This may include computers, software, machining tools, or testing and calibration equipment.

The lost income included in this estimate is based on a three month total down-time. This time could be lengthened by shipping delays from manufacturers, components that do not fit, or licensing delays. Because the replacement of primary loop components can be done without a long shutdown, the schedule is set by the time needed to upgrade the secondary system (piping and cooling towers). A logical work schedule and an effective QA program will be implemented to reduce the reactor down-time. For example, the reactor should continue to run until the replacement components have all arrived and passed an in-house QA inspection.

The payroll of the reactor laboratory must be paid while any refurbishments are being done. Payroll and Operation Costs include the salaries of most NRL employees, permits, services, and insurance. The estimated operating cost chargeable to the relicensing is based on a three month time period.

The costs of site preparation and cleanup are included in the cost of the major components found in Table 5.4-3.

5.4.3 Other Direct Costs

In order to relicense MITR-II at its current power level of 5 MW for another twenty years, many key components will need to be replaced. Preliminary design work done over the past two years has provided the technical specifications necessary to make rough estimates of the cost. The estimated direct costs are shown in Table 5.4-3.

Table 5.4-3
Other Direct Costs For Relicensing MITR-II

DIRECT COST CATEGORY	COST (1994 DOLLARS)
Primary Heat Exchangers	\$32,800
Primary Pumps	\$92,800
Secondary Pumps	\$92,800
Secondary Piping	\$90,100
Cooling Towers & Filtration System	\$210,100
Control System	\$47,500
Containment	\$139,100
Effluent Release Stack	\$5,000
Electrical & Instruments	\$280,000
Shield Cooling	\$25,000
Decontamination/Waste Disposal	\$95,000
TOTAL	\$1,110,200

* Complete waste disposal costs cannot be estimated until a low-level waste site can be designated.

The three main heat exchangers currently in place in MITR-II will need to be replaced with newer designs. The old heat exchangers are inadequate to remove 5 MW of heat during days with high wet-bulb temperatures, forcing the reactor to reduce power on hot days. A number of tubes have been plugged to isolate leaks in the existing system and these heat exchangers would contribute to unreliable operation if not replaced.

This updated analysis uses cost estimates obtained from Alpha-Laval. The Alpha-Laval models can remove the same amount of heat as the shell and tube heat exchangers that are currently used, while taking up half the space and costing much less. A study by M. Arno suggests that plate and frame heat exchangers are relatively easy to disassemble for routine maintenance, and that there is no cross-leakage problem between the primary and secondary sides [20]. Operational experience at the National Bureau of Standards and Technology (NIST) has shown that any leaks that may develop would be external rather than between fluid systems. As in the current arrangement, three heat exchangers, each to 50% capacity, will be installed. NIST has reported some difficulty with fouling on the secondary side, so a new secondary filtration system may need to be installed.

The primary and secondary system pumps (two of each) were installed in the 1960s. They are showing deteriorating performance and are becoming increasingly difficult to maintain due to an unavailability of spare parts. Estimates for replacement pumps have been received, based on the required pump heads and flow rates. The possible use of magnetic bearings is being studied to see if the improved performance and ease of maintenance would be advantageous to the project.

The underground secondary piping and electrical lines from the containment building to the cooling towers needs replacement. These pipes have shown some leaks and are inaccessible in their current configuration. The current pipes to and from the cooling towers will be removed and replaced. The same channel will be used to run a conduit with

new electrical wires for power and control to the towers. It would be desirable to cover this channel with a removable lid to allow access for maintenance and repair work to be carried out without having to dig the pipes up again. This would require a set of pipe heaters to be available in the winter.

The two wooden cooling towers that are currently in use were installed 35 years ago. Severe weather and aging has led to deterioration of the structural components and reduced performance. Marley Cooling Tower Co. has provided the MIT Nuclear Reactor Laboratory (NRL) with their recommendation for a replacement system. This recommendation is based on the restricted physical envelope (due to the current tower basin size), operating temperature range, and secondary flow rate provided by the NRL. The Marley system is a set of two modular heat exchangers and associated fans and piping [20]. Also included in the cooling tower estimate is the cost of installing a filtration system and a fire alarm system.

The control blade drive motors that are used to raise the six control blades and one regulator rod are old and have no commercially available spare parts. These motors need to be replaced with similar models. The control system itself will be upgraded.

The containment building is pressure-tested periodically for leaks and spot-checked for corrosion near the ground. A recently renewed anode system is being used to protect against future corrosion. The truck lock will require considerable work to allow it to be usable. Two gaskets, a seating surface, and structural portions of the lock need to be repaired. This cost estimate assumes that the containment will be inspected and any corrosion damage will be repaired, and that the truck lock will be made serviceable.

The electrical system components, including emergency batteries, switch boxes, wiring, transformer, and generator, need refurbishment or replacement. Only rough estimates are available for the large components because it is unclear what the electrical requirements will be for the reactor system with its replacement components. The motor control centers are not included in this cost analysis.

The current shield coolant system will be refurbished to repair one of the coolant coils in the shield. Also, the shield coolant loop components may need to be refurbished or replaced.

Decontamination and waste disposal costs for all components that need to be replaced are included together. Secondary side components (pumps, piping, etc.) can be disposed of by conventional means. The cooling towers will be dismantled and the wood will be sold as scrap. The heat exchangers and primary side components will need to be evaluated for decontamination prior to disposal. Until a low level waste depository will accept these shipments, it is not possible to assign a disposal cost to these components. Adequate on site storage space is available while a low level waste disposal site is under development in Massachusetts.

5.4.4 Summary of Costs To Relicense MITR-II At 5 MW

Based on the assumption that a relicensed reactor will be operated for at least 20 years, the necessary design, operating, and direct component costs have been estimated as described above. As previously noted, these costs are sensitive to several variables:

- 1) The component replacement schedule will determine the required reactor down time. Delays from manufacturers and installation problems can be minimized with a thorough quality assurance program. Licensing and start-up delays are more difficult to control. Component replacement would not start until license authorization is obtained.
- 2) The prices for major components could change if system parameters, due to licensing requirements, are different than those currently assumed. In addition, these costs are based on estimates from a small number of sources. Once a design is final, it will be possible to obtain competitive bids on these components.
- 3) Before or during the refurbishment other systems or components could be found to be in need of replacement.

This analysis includes "best estimate" costs for all work outlined above. A 25% contingency factor has been included to account for unforeseen work or price overruns. The following systems have been assumed to be adequate for future operation, and have not been included in this analysis:

- 1) Shield Coolant System
- 2) Coolant Purification System and Ion Columns
- 3) Core and Reflector Tanks
- 4) Primary Piping System
- 5) Spent Fuel Pool
- 6) Emergency Core Cooling System (ECCS)
- 7) Graphite Reflector
- 8) Reflector Coolant System
- 9) Control System (except motors)

Table 5.4-4 shows a summary of the estimated costs for relicensing at 5 MW.

Table 5.4-4
Summary of Costs For Relicensing MITR-II At 5 MW

COST CATEGORY	COST (1994 DOLLARS)
Engineering and Design	\$306,000
Operating Costs	\$1,158,000
Other Direct Costs	\$1,110,200
Contingency	\$643,500
TOTAL	\$3,217,700

APPENDICES

The following appendices (A through J) summarize the studies which are relevant to both relicensing at 5 MW and upgrading to 10 MW.

Appendix A: MITR-III TRIGA Design [23]

A.1 Introduction

In considering new core designs, a TRIGA-fueled core was selected as one of the potential upgrades of the MITR-II. Therefore, studies were done to investigate the thermal hydraulic and neutronic performance of a proposed TRIGA design to determine the design's ability to fulfill user requirements. The neutronic and thermal hydraulic analyses were assisted by the use of the computer codes MCNP [10] and PARET [13], respectively. In addition, the program MathCAD was used to calculate in-core boiling conditions and the pressure drop in the primary cooling system.

The TRIGA fuel is a commercially available research reactor fuel with enhanced inherent safety characteristics. These advantages make the TRIGA fuel an attractive option for the MITR-III.

A.2 MITR-III TRIGA Core and Fuel Design

The following two premises were adopted and guided all core design considerations:

- 1) Use of existing MITR-II core tank
- 2) Use of LEU TRIGA fuel (rod type) developed by General Atomics (GA)

The reasons for using these two constraints in the design are for enhanced safety, and to maintain an economical design. The LEU fuel was chosen according to NRC requirements and the TRIGA fuel was selected for its high negative fuel temperature coefficient of reactivity.

A.2.1 TRIGA fuel Description

The TRIGA fuel uses a Uranium-Zirconium-Hydride-Erbium fuel material in which the hydrogen moderator and the erbium absorber are homogeneously contained within the fuel material. The mixture of fuel, moderator and absorber provides an inherent safety feature - a prompt negative fuel temperature coefficient of reactivity - which is the major advantage of using TRIGA fuel. TRIGA fuel pins are now available in two sizes, 1/2" and 3/8" diameter, and both types were considered in this study. Tables A-1 and A-2 show specifications for different types of fuel pins.

Table A-1
Reactivity Coefficients and Parameters [24]

	Λ (μ s)	β_{eff}	Coolant Temperature Coefficient (\$/°C)	Void/Density Coefficient (\$/%void)	Doppler Coefficient (\$/°C)
HEU	55.96	7.607E-3	1.537E-2	0.3257	3.6E-5
LEU	43.74	7.275E-3	1.082E-2	0.4047	3.31E-3

Table A-2
Nominal Specifications for TRIGA LEU Fuel Pin

Fuel composition	U-ZrH-Er
Uranium Content	45 wt %
Uranium-235 enrichment	19.7 wt % (nominal)
Hydrogen-to-zirconium ratio	1.6
Total length	610 mm
Effective length	560 mm
Fuel OD (unclad)	8.53 mm
Clad OD	9.14 mm
Cladding material	Incoloy 800
Clad Thickness	0.3 mm

A.2.2 MITR-III Core Designs

Several MITR-III core designs using TRIGA fuel were analyzed: one set of designs was based on the basic model proposed by GA [24] using 1/2" TRIGA fuel while the other set used 3/8" fuel. Figure A-1 shows a schematic of the GA-proposed design, and the details of the hexagonal fuel cluster and the fuel rod are given in Figures A-2 and A-3 respectively. In addition to the 19-rod fuel cluster, GA also manufactures a 37-rod hexagonal cluster using the 3/8" fuel rod. Figures A-4 and A-5 give schematic views of the fuel rod and the cluster respectively. This fuel cluster was also used to model MITR-III cores in this study.

Figure A-1. MITR Core Configuration with TRIGA 19-Rod Hexagonal Fuel Clusters

Figure A-2. Compact 19-Rod TRIGA Hexagonal Cluster

INCOLOY CLADDING: 0.542 O.D.

0.016 THICK

COMPRESSION SPRING

FUEL PELLETS / 22.00 TOTAL LENGTH

2.12

4.00

30.13

DIMENSIONS IN INCHES

Figure A-3. TRIGA Fuel Rod for 19-Rod Cluster

Figure A-4. TRIGA Fuel Rod for 37-Rod Cluster (as manufactured by GA)

SCALE: 1:1

Figure A-5. TRIGA 37-Rod Hexagonal Fuel Cluster

A.2.3 Calculation Method

The computer code used for neutronic analysis in this study was MCNP ver. 4x-c, which is a continuous energy Monte Carlo code with explicit modeling features. One of the advantages in using this code is that neutron cross section processing is not required. However, the MCNP data library did not contain any cross sections for erbium, and after comparison of resonance integrals and thermal cross sections, protactinium-231 was selected as a substitute.

All of the MCNP cases were run for a minimum of 450,000 neutron histories to decrease associated statistical errors. These errors are given at one standard deviation around the mean value.

A.2.4 Description of Core Model

All of the designs were established for 10 MW operation. Fluxes reported are relative to operation at 10 MW. One of the main premises of this study was the use of the existing MITR-II core tank, and therefore, all core configurations were based on the schematic shown in Figure A-6. An existing detailed MCNP model of MITR-II was simplified and used to model out-of-core regions for the MITR-III design [25]. The only variables in the non-fuel region were the thickness of the bottom core plate and the presence of a beryllium reflector region. After initial calculations, the thickness of both of these elements was set to 2 cm. Furthermore, the length of the fuel rods was decreased to a total of 61 cm because the unfueled region at the top of the rods did not serve any purpose in the MITR-III design.

A.3 Results of Neutronic Analysis

During the course of this study, it was found that when LEU TRIGA fuel was used in various core configurations, the core could not be made critical because of the presence of burnable poison. Calculations showed that this burnable poison contributed between 7 and 7.5% $\Delta k/k$ of negative reactivity to the system, depending on whether the 19-rod or the 37-rod fuel clusters were used. As a consequence, no burnable poison was used in subsequent models for initial design purposes. Once the design could be established, some burnable poison (erbium) and its attendant safety enhancement would be added. Moreover, all the neutronic results were calculated for cold coolant and cold clean fuel conditions.

A.3.1 Cores Using 19-rod Fuel Cluster

The k_{eff} for this model was calculated to be 1.051 ± 0.002 , which meant that the core would be subcritical in the presence of 1.8 w/o erbium. In an attempt to increase the excess reactivity, the fuel size was varied with a constant pin pitch to determine if changing the moderator-to-fuel ratio would be helpful. However, the reference case fuel cluster was already undermoderated and variations in fuel size were not helpful. To increase the excess reactivity, the water in the downcomer region was replaced with

beryllium reflector, and this raised the k_{eff} of the model to 1.092 ± 0.002 . Hence, this model, albeit unrealistic, had an excess reactivity of 2 % in the presence of burnable poison.

A.3.2 Cores Using 37-rod Fuel Cluster

After the unacceptably low k_{eff} values for models using the 19-rod cluster, the fuel type was switched to the 37-rod cluster as shown in Figure A-7. The k_{eff} for this specific unpoisoned model was found to be 1.029 ± 0.003 . The reason for the lower k_{eff} is that the 37-rod cluster contains less fuel than the 19-rod cluster. Since the size of the core tank could not be increased, more fuel was added by replacing twelve of the in-core beryllium blocks with partial assemblies, as shown in Figure A-8. This model contained almost as much fissile material as the 19-rod cluster model. The resulting k_{eff} was raised to 1.047 ± 0.002 . Since this core would also be subcritical in the presence of burnable poison, the moderator-to-fuel ratio was varied by changing the fuel size while maintaining a constant pin pitch. However, the resulting increase in k_{eff} was negligible.

A.3.3 Final MITR-III TRIGA Core

The final design consisted of 16 full and 12 partial (1/3 of full size) 37-rod fuel clusters and a 2 cm beryllium reflector. It was also determined that if the burnable poison content were reduced to 0.6 weight percent, this core would still retain an excess reactivity of 0.4% $\Delta k/k$. The axial neutron flux profile was determined for the in-core irradiation facilities and the cylindrical region below the core. These fluxes were volume averaged over one-centimeter segments for the two regions. In the irradiation facility, consisting of aluminum and 7.34 cm from core centerline, the peak fast flux was slightly higher than 2.18×10^{14} neutrons/cm²-sec at 10 MW operation. However, the flux profile at all energies was not as skewed towards the bottom as was expected. In the D₂O filled region below the core and between the beam tubes, the peak thermal flux was found to be approximately 1.05×10^{14} neutrons/cm²-sec.

A.3.4 Conclusion and Recommendations

A TRIGA MITR-III core design could be made critical with 0.4% $\Delta k/k$ excess reactivity at BOL, with the use of 12 partial and 16 full 37-rod hexagonal fuel clusters, and the content of burnable poison reduced to 0.6 weight percent. However, the system would be subcritical if standard LEU TRIGA fuel was used and therefore, it was concluded that the use of LEU TRIGA fuel within the 20" core tank is of only limited capability. If the MITR-II core tank is to be retained, the fuel enrichment should be increased, and if the LEU TRIGA fuel is selected, then the 20" core tank would have to be enlarged to add at least another ring of fuel clusters.

Scale 1:5

Figure A-6. Schematic of MITR-III Core Configuration

Figure A-7. MCNP Core Model Using 37-Rod Fuel Cluster

Figure A-8. MCNP Core Model Using 37-Rod Fuel Cluster and Partial Assemblies

A.4 Thermal Hydraulic Analysis Using PARET

The PARET code used in the thermal hydraulic analysis was originally developed for the analysis of the SPERT-III experiments at the Idaho National Engineering Laboratory for temperatures and pressures typical of power reactors. It was then modified to include a selection of flow instability, departure from nucleate boiling, single and two-phase heat transfer correlations, and a properties library considered more applicable to the low pressures, temperatures and flow rates encountered in research reactors. It provides a coupled thermal hydraulic and point kinetics capability with continuous reactivity feedback, and an optional voiding model that estimates the voiding produced by subcooled boiling. The code has also been used to analyze the International Atomic Energy Agency (IAEA) 10 MW benchmark cores with both high enrichment uranium (HEU) and low enrichment uranium (LEU) fuel for protected and unprotected (no scram) transients [26].

A.4.1 Current Design of MITR-III TRIGA Core

A TRIGA core design, using 37 pins 3/8" TRIGA fuel in 16 hexagonal assemblies and 12 partial assemblies was chosen due to neutronic considerations. The core consists of a total of 664 fuel pins with a pin pitch of 1.12 cm. The core's hydraulic diameter is 0.91 cm with a total flow area of 0.04295 m².

The axial heat flux distribution used in the thermal hydraulic calculations was adopted from GA's calculated result for the MITR-III core design. It was assumed that flux shape remained the same since the D₂O and graphite reflectors, being the major contributors to the bottom-peaked flux shape, are common to both designs. General Atomic's results were curve-fitted and the following normalized heat flux was obtained:

$$\overline{f(z)} = 3.936 z^3 - 9.685 z^2 + 3.239 z + 0.933 \quad (\text{Eqs. A-1})$$

A calculation was carried out with this heat flux distribution and a hot channel factor of 1.8 to observe the flow rate effect on heat removal. The results show that nucleate boiling occurs when the mass flux is below 4100 kg/m²-sec (2810 gpm). This flow rate could be further reduced if the in-core thermal hydraulic limit was critical heat flux (CHF) instead of boiling. In-core boiling is tolerable for TRIGA fuel, due to the high melting point of Incoloy 800 (1360°C). Therefore, the focus of the thermal hydraulic analysis was moved to nucleate boiling.

A.4.2 Prediction of In-Core Boiling and Pressure Drop

Saha/Zuber's correlation was used to determine onset of bubble departure [27]. Levy's equation [28] was used to determine flow quality as a function of local thermal equilibrium quality and the thermal equilibrium quality at the bubble detachment point. The homogenous equilibrium model (HEM) was then used to determine the pressure drop for two-phase flow.

A.4.3 Critical Heat Flux (CHF)

The critical heat flux calculation was carried out using GA's CHF correlation [29]. It was found that the departure from nucleate boiling occurred in the narrow gaps between the hottest rods. Experimental results showed that there was no vapor generation except in the narrowed gap. Also, it was found that the spacing of the fuel pins is one of the controlling parameters for prediction [29].

In the CHF calculations, the heat flux is assumed bottom-peaked as described earlier and a hot channel factor of 1.8 was used for the hottest assembly. The suggested CHF is 1.65 for TRIGA fuel [24]. Therefore, according to the results, the suggested operating mass flow rate is 141.7 kg/s, corresponding to 2260 gpm. The core inlet and outlet temperatures are 38 and 68°C, respectively. The following primary loop pressure drop calculation, thus the pumping power prediction, is based on this flow rate.

A.4.4 Primary Loop Pressure Drop and Pumping Power Calculation

The primary system was assumed to be the same as that of MITR-II. Pressure drop for the out-of-core region of the primary cooling system is calculated using the following equation:

$$\Delta P = \Delta P_f + \Delta P_a + \Delta P_g + \Delta P_{form} \quad (\text{Eqs. A-2})$$

where

$$\Delta P_f = f_{fo} \frac{L_x}{D_*} \left(\frac{G^2}{2\rho_f} \right) ; \quad \Delta P_g = \rho_f g L_z ;$$

$$\Delta P_{form} = K_{form} \frac{G^2}{2\rho_f} ; \text{ and } \Delta P_a = 0.$$

L_x is the distance the coolant travels in a certain component, and L_z is the elevation of this component. The liquid densities in the hot leg and cold leg are calculated according to the core outlet and inlet temperatures. The coolant temperature in the heat exchanger is assumed to be the average of the inlet and outlet temperatures. The resulting pressure drops are shown in Table A-3. An area change at the entrance of cold leg 3 results in a large pressure drop of 121.7 kPa. Heat exchanger and cold leg 1 produce large pressure drops due to the small equivalent diameter and large gravity pressure drop, respectively. Pumping power was calculated using an energy balance equation:

$$P_{pump} = \frac{m \Delta P_{total}}{\rho_f} \quad (\text{Eqs. A-3})$$

The calculated pumping power for the suggested flow rate is 41.0 kW (55.0 hp). If the pump efficiency is assumed to be 80%, then the power required will be 51.3 kW (68.8 hp).

Table A-3
Pressure drop of the primary cooling system for the suggested flow rate

Component	Pressure Drop (kPa)
Core	37.1
Mixing Area	19.95
Hot Leg	-18.75
Heat Exchanger	47.15
Cold Leg 0	97.43
Cold Leg 1	-11.86
Cold Leg 2	-6.27
Cold Leg 3	121.7
Cold Leg 4	-3.19
Total pressure drop = 283.2 kPa	

A.4.5 Summary and Recommendations

A thermal-hydraulic analysis was carried out for the proposed TRIGA core selected by neutronic considerations. Both PARET code and a MathCAD program were used for the thermal hydraulic analysis. The results showed that the operating limit of the TRIGA design is the critical heat flux instead of parallel channel flow instability. Table A-4 is a summary of the proposed design with some of its thermal hydraulic characteristics.

Table A-4
Summary of the thermal-hydraulic analysis

Flow rate	141.7 kg/s (2260 gpm)
(ONB occurs at 176.1 kg/s)	
MCHFR	1.68
Primary Loop Pressure Drop	283.2 kPa
Pumping Power	51.3 kW (68.8 hp)
Inlet/Outlet Temperatures	38°C / 68°C

The following recommendations are made for further study of TRIGA MITR-III design:

- (1) Boiling incipience may be too rigorous a constraint for TRIGA MITR-III core design. Further study is needed to investigate the effect of wall voidage (partial subcooled) region on both flow instability and critical heat flux.
- (2) The HEM model should be replaced by the drift model or a two-fluid model for a subcooled boiling calculation.

- (3) The calculated pressure drop in cold leg 3 is large, and is mostly due to the effect of a sudden area change. Therefore, the loss coefficient should be investigated further since this term has a dominant effect in the pressure drop calculation.
- (4) Transient analyses should be performed to establish the safety limits and limiting safety system settings.

A.5 Overall Summary

A TRIGA-Fuel core can be designed to fit into the present MITR-II core tank and operate satisfactorily from a thermal-hydraulic point of view. However, these investigations show that the neutronic capability is too limited for reasonable operation at 10 MW. A new slightly larger core tank would allow the desired flexibility of operation for the MITR-III.

B.1 Introduction to SILOE Reactor

The SILOE research reactor is a French pool type reactor. Its characteristics are summarized in Table B-1. The reactor is cooled by light water, and reflected by beryllium elements surrounding the core. Each fuel element contains 23 fuel plates, while the dummy elements contain only 12 fuel plates to allow for the placement of two 3.6 cm (1.4 in) diameter experimental orifices. Control is provided by control elements made up of 17 fuel plates and 2 hafnium blades. The 35 MW SILOE reactor has a thermal flux of 6×10^{14} n/cm²s in the H₂O reflector and a fast flux of 4.5×10^{14} n/cm²s in the experimental facilities.

The purpose of this study is to analyze the feasibility of using a SILOE-type core in the MIT Research Reactor. The following sections outline the modifications needed to transfer the SILOE design to the MITR and the expected reactor performance.

Table B-1
Characteristics of French SILOE Reactor

PARAMETER	SILOE
Thermal Power	35 MW
Moderator	H ₂ O
Reflector	Beryllium, H ₂ O
Fuel	U ₃ Si ₂ -Aluminum Element (LEU)*
Core Configuration	22 Fuel Elements 6 Dummy Elements 5 Control Elements (10 hafnium blades) Total of 663 Fuel Blades
Primary Coolant Flow Rate	9400 gpm (2150 m ³ /hr)
Coolant Temperature	
Inlet:	40°C
Outlet:	49°C
Cycle Lifetime	21 Days

*For the MITR application 19.75 % (LEU) was used.

B.2 General Design for MITRB.2.1 Fuel Elements

The fuel element designed for use in the MITR is composed of 23 fuel plates containing 19.75 % enriched uranium. The fueled portion of each plate is 6.23 cm x 0.051 cm x 59.65 cm (2.45 in x 0.02 in x 23.5 in). The uranium density varies from 4.0 g/cm³ in the four external plates to 4.8 g/cm³ for the other plates. The fuel elements are clad

with 0.038 cm (0.015 in) of thick aluminum cladding. A gap of 0.21 cm (0.018 in) is provided between the plates to allow for the flow of coolant.

The fuel element design includes a 0.8 cm (0.19 in) diameter safety hole to allow for some coolant flow in the event of a nozzle blockage. In order to accommodate the MITR-III user requirements, the unfueled regions of the fuel elements have been shortened, improving the flux below the core. As in the MITR-II, the fuel elements have been designed to be symmetric to allow for an even burnup by inverting the elements.

B.2.2 Dummy Elements

The dummy elements have the same shape as the fuel elements but are filled with aluminum instead of fuel plates. To allow for the placement of in-core experiments, a 7.4 cm (2.9 in) diameter hole is included in the middle of each dummy element. For the MCNP calculations, the hole is filled with aluminum to simulate the presence of experimental equipment.

B.3 Core Design

B.3.1 Control System

The control system includes three hafnium control blades (28.9 cm x 2 cm x 71 cm) and two hafnium regulating rods (3.64 cm diameter x 71 cm long) located around the perimeter of the core.

B.3.2 Core Configuration

The core is made up of 25 fuel elements and three dummy elements arranged as shown in Figure B-1. In this figure, the control blades and regulating rods can be seen outside the core.

B.4 Reflector Tank

A new reflector tank has been designed in order to optimize the usefulness of the SILOE design. The new reflector tank includes a 10 cm diameter cold neutron source with liquid hydrogen (10 K) with a new beam port attached to it. Also included is a new medical therapy room on the side of the core, with a D₂O shutter tank to select epithermal neutrons. The new reflector design minimizes the amount of heavy water below the core tank (only 12.5 cm between the core tank and the blister), and uses a larger D₂O blister between the core bottom and the lower therapy room.

B.5 Neutronics Study

C.5.1 Criticality

Using MCNP to model the SILOE core, the eigenvalues shown in Table B-2 were obtained.

Table B-2
Eigenvalues for SILOE Core in the MITR

CONDITIONS	EFFECTIVE MULTIPLICATION (k_{eff})	ERROR (%)
Control Blades Up	1.04	0.13
Control Blades Down	0.97	0.13
D ₂ O Shutter Under The Core Empty	1.04	0.13
D ₂ O Shutter Next To The Core Empty	1.02	0.19

Figure B-1 SILOE Core Configuration

B.5.2 Flux distribution

The flux distribution found by using MCNP was divided into the following ranges for analysis:

- Thermal flux : $E < 0.5$ eV
- Epithermal flux : $0.5 \text{ eV} < E < 0.1 \text{ MeV}$
- Fast flux : $0.1 \text{ MeV} < E < 20 \text{ MeV}$

Fluxes in the Core

Figures B-2 and B-3 show the flux distributions in the dummy element and the fuel element, respectively. The maximum thermal flux found in the dummy element is $4.24 \times 10^{13} \text{ n/cm}^2\text{s}$, and the maximum fast flux is $1.63 \times 10^{14} \text{ n/cm}^2\text{s}$. The peak fluxes in the fuel element are: 4.15×10^{13} thermal $\text{n/cm}^2\text{s}$ and 1.64×10^{14} fast $\text{n/cm}^2\text{s}$.

Fluxes In the Reflector

The fluxes in the reflector near the new D₂O shutter (located on the side of the core tank) are shown in Figure B-4 (D₂O shutter full) and Figure B-5 (D₂O shutter empty).

The flux at the entrance of the beam tube that leads from the cold neutron source is shown in Figure B-6. In this plot, the thermal flux is divided into three energy groups (0.001 to 0.01 eV, 0.01 to 0.1 eV, and 0.1 to 0.5 eV).

Figure B-7 shows a diagram of the model used to find the fluxes below the core. The fluxes in Region 2 are shown in Figures B-8 and B-9 (for the cases of a full and empty D₂O shutter, respectively). Table B-3 lists the fluxes in the other regions labeled with roman numerals in Figure B-7.

Table B-3
Flux in Regions Below Core

Region	Thermal Flux ($\text{n/cm}^2\text{-s}$)	Epithermal Flux ($\text{n/cm}^2\text{-s}$)	Fast Flux ($\text{n/cm}^2\text{-s}$)
Cell I	5.65E13	2.56E13	1.47E13
Cell II	---	See Figures	---
Cell III	1.15E13	1.80E12	3.56E11
Cell IV	8.51E12	9.80E11	1.96E11
Cell V	1.44E12	1.24E11	3.59E10
Cell VI	1.19E12	1.03E11	2.66E10

..... Fast Flux ($0.1 \text{ MeV} < E$)
 — Epithermal Flux ($0.5 \text{ eV} < E < 0.1 \text{ MeV}$)
 - - Thermal Flux ($0.01 \text{ eV} < E < 0.1 \text{ eV}$)

Figure B-2. Flux in Dummy Element

Figure B-3. Flux in Fuel Element

..... Fast Flux ($0.1 \text{ MeV} < E$)
 — Epithermal Flux ($0.5 \text{ eV} < E < 0.1 \text{ MeV}$)
 - - Thermal Flux ($0.01 \text{ eV} < E < 0.1 \text{ eV}$)

Figure B-4. Flux in New D₂O Shutter
(on side of core, shutter full)

Figure B-5. Flux in New D₂O Shutter
(on side of core, shutter empty)

Figure B-6. Flux in Beam Tube Next to the Cold Neutron Source

Figure B-7. Diagram of MCNP Model Below the Core

Figure B-8. Flux in D₂O Shutter Below the Core (Shutter Full)

Figure B-9. Flux in D₂O Shutter Below the Core (Shutter Empty)

B.6 Thermal Hydraulics

The first step in the thermal-hydraulic study was to determine the thermal distribution in the core with the reactor design outlined above.

The second step was to investigate the Safety Limit and the Limiting Safety System Setting for the following situations:

- Steady state.
- Transient: loss of coolant flow.
- Accident: blockage of the hot fuel element nozzle and blockage of the hot channel bottom.

The thermal-hydraulic conditions under these situations were analyzed using the PARET code.

B.6.1 Thermal Distribution

The first step of the thermal-hydraulic study was to determine the heat distribution within the system. In doing so, several simplifying assumptions have been made:

- Gamma rays and electrons are not included in the heat distribution calculation.
- The axial heat distribution is modeled by a discrete fuel plate model.
- The radial heat distribution is calculated for each fuel element.

Figure B-10 shows the radial distribution used in the calculations. Because the PARET code requires an axial distribution to be provided as well, a two-channel model, using a hot and an average channel, has been chosen.

Figure B-10. Radial Power Distribution in SILOE Core

A calculation of the thermal distribution for each fuel plate of an average fuel element, combined with the radial distribution, gives an estimate of the hot channel conditions. In this case the peaking factor in the hot channel is only 1.72, whereas in the SILOE thermal-hydraulic model, the value of 2.4 was obtained. Although this value seems low, it will be used in this analysis. In order to obtain conservative results, the hot channel area has been assumed to be 10 % of the total channel area. This thermal-hydraulic model is shown in Figure B-11.

Figure B-11 Axial Power Distribution of SILOE Core

The power peaking at the bottom and the top of each channel is due to the presence of the heavy water shutter and the light water pool. These results are similar to the results for the current MITR-II Reactor.

The average heat flux in the core is 0.22 MW/m^2 , whereas if the MITR-II was operated at 10 MW, it is estimated to be 0.44 MW/m^2 . The power density in the SILOE-type core is 107 kW/l, compared to 175 kW/l estimated for MITR-II.

The average heat flux and power density values are quite low, indicating that the SILOE core will be limited by neutronics, not by thermal-hydraulics.

B.6.2 Steady State

Several steady-state calculations were performed. The parameters of interest included: the maximum outlet coolant temperature, the average outlet coolant temperature, the maximum cladding temperature, and the maximum fuel temperature.

The inlet coolant temperature is assumed to be 40°C. This temperature is subject to some variation. If there was any problem in the primary or secondary coolant systems, this temperature could vary.

This study has been conducted with three coolant flow rates. The current flow rate in the MITR-II is approximately 2000 gpm. An estimated flow rate for the MITR-III (at 10 MW) is 2400 gpm. Finally, a flow of 4000 gpm was considered because it gives results which are most similar to the SILOE reactor in France. Table B-4 shows the results of the steady state calculations.

Table B-4
Steady State Results at Various Flow Rates

Coolant Flow Rate (gpm)	Average Outlet Coolant Temp.(°C)	Coolant Maximum Temp.(°C)	Cladding Surface Maximum Temp.(°C)	Fuel Surface Maximum Temp.(°C)	Fuel Centerline Maximum Temp.(°C)	Average Reynolds Number
2000	59.2	64.4	85.6	86.3	86.7	12,400
2400	56	60.3	79.6	80.3	80.6	15,100
4000	49.6	52.2	66.4	67.1	67.4	22,750

In the current MITR design, thermal-hydraulics is the limiting factor. To allow for operation of the MITR-III at 10 MW, the primary coolant flow rate will be increased. Recent studies estimate that a flow increase of at least 20 % will be required. Therefore, the following thermal-hydraulic transient and accident analyses were done with a coolant flow rate of 2400 gpm.

B.6.3 Loss of Flow Transient

In case of a loss of flow accident, the current MITR design can remove all the decay energy by natural convection. One of the purposes of this study is to determine if the SILOE design permits such a decay heat removal mechanism without reaching the safety limits.

Prior to the loss of coolant flow accident, the reactor is assumed to be in steady state. Once the transient begins, the flow temporal behavior in the core was approximated by the following empirical relation:

$$\frac{G(t)}{G(0)} = 2.283e^{-0.1t} + 0.0283t - 1.283 \quad (\text{Eqs. B-1})$$

where

$G(t)$ = mass flux at time t ($\text{kg/m}^2\text{-s}$)

$G(0)$ = mass flux at the beginning of the accident ($\text{kg/m}^2\text{-s}$)

t = time (between 0 and 7.6 seconds)

Although this equation is derived from experiments performed at the MITR-II, it was used to simulate a loss of coolant flow accident for the proposed SILOE design. When the new coolant system is installed, further study of the coolant flow rate time behavior in a loss of flow accident will be necessary.

The flow trip set point is at 85 % of the nominal flow. The detection delay time is 0.5 seconds. This value will give conservative results. The total negative reactivity insertion during the scram is assumed to be 9\$ (value given by the result of the neutronics study). The blade drop speed is assumed to be 1.2 m/s. Figure B-12 shows the maximum coolant and cladding temperatures as functions of time.

Figure B-12. Temperature Distributions During Loss of Flow

The flow trip is detected after 0.8 seconds. The maximum coolant temperature reaches 63.5°C and the maximum cladding temperature reaches 84°C. In this condition, Onset of Nucleate Boiling does not appear, and fuel damage does not occur. After 1.7

seconds, both temperatures go down due to the reactor scram. After 4 seconds, these temperatures go up again due to the reduced coolant flow. After 16 seconds, they peak (at 79.5°C and 75.5°C, respectively), and then decrease thereafter.

This study shows that if this accident occurred, the Onset of Nucleate Boiling would not occur because of the low maximum cladding temperature. Therefore, there would not be any fuel damage. Moreover, it shows that the decay heat could be removed by natural convection.

B.6.4 Accident Analyses

Blockage of the Hot Element Nozzle

The Blockage-of-Element-Nozzle accident has not been investigated for the MITR-II Safety Analysis Report. At the bottom of the fuel element, a lateral hole permits coolant circulation in case of element nozzle blockage. The design of this lateral hole determines the coolant flow rate if the accident occurs. This study has been done using PARET, and only the steady state has been investigated.

The main assumption used is that the pressure drop is equal in the blocked and unblocked elements. The steady state simulation gave a pressure drop of 12 kPa. The friction coefficient is assumed to be 0.05. The outlet pressure drop coefficient is negligible compared to the inlet pressure drop coefficient. The average coolant outlet temperature was calculated to be 70.7°C (compared to a steady state temperature of 56.0°C), with a maximum coolant temperature of 79.7°C. The cladding reaches a maximum temperature of 105.8°C.

Blockage of Channel Bottom

The Blockage-of-Channel-Bottom accident has been investigated and is the design basis accident for the current MITR. According to the literature, there are two types of burnout: limited circulation burnout and flooding limited burnout. The limiting burnout is limited circulation burnout in the SILOE design, due to the large size of the coolant channel.

As a result, the complete voiding of a blocked channel is not possible. However, if the vapor quality at the exit of the channel approaches unity, it could lead to dry spots on the heated wall. These dry spots could cause Departure from Nucleate Boiling, and possibly fuel damage.

The following correlation gives the heat flux for Departure from Nucleate Boiling for a blocked channel bottom:

$$q_{DNB}'' = \left(\frac{A_f}{A_h} \right) h_v [\rho_v (\rho_l - \rho_v) g L]^{1/2} \quad (\text{Eqs. B-2})$$

where

- q_{DNB} = heat flux for DNB (W/m^2)
- A_f = coolant area (m^2)
- A_h = heated area (m^2)
- h_{lv} = latent heat of vaporization (kJ/kg)
- ρ_v = vapor density (kg/m^3)
- ρ_l = liquid density (kg/m^3)
- g = acceleration of gravity (m/s^2)
- L = heated length (m)

Note: The coolant properties are evaluated at the saturation temperature.

The heat flux for Departure from Nucleate Boiling is 0.57 MW/m^2 . Since the average heat flux is 0.22 MW/m^2 and the peak factor is 1.72 (so the maximum heat flux is 0.41 MW/m^2), the heat flux for Departure from Nucleate Boiling is not reached. There will be nucleate boiling in the blocked channel, because of the decreasing flow rate. But even with boiling in the channel, fuel damage is not expected.

B.7 Conclusion

The following table (Table B-5) summarizes the differences between the SILOE approach and the current MITR-II upgraded to 10 MW for the same mass flow rate of 2400 gpm.

Table B-5
Comparison of SILOE and MITR-II at 10 MW

PARAMETER	SILOE	MITR-II at 10 MW
Mass Flux ($kg/m^2\text{-s}$)	1800	3400
Coolant Average Speed (m/s)	1.9	3.4
Coolant Area (cm^2)	800	430
Average heat flux (MW/m^2)	0.22	0.44
Maximum heat flux (MW/m^2)	0.4	1.3
Power density in the core (kW/liter)	107	175

Appendix C: Methodology for Transient Analysis [6]

The following sections summarize the modeling of the MITR system as used in the development of the simulation computer code TAMLOC.

C.1 Energy and Mass Conservation

Figure C-1 illustrates the representation of the MITR system in TAMLOC. The primary coolant system was divided into eight control volumes for which conservation of mass and energy laws were applied. The control volumes included the core, three regions above the core, the hot leg piping (including the pumps), two heat exchangers, and the cold leg piping (including the downcomer).

Figure C-1. Control Volumes for the MITR Thermal Model

The physical model for these components employs the following assumptions, such that a coarse-mesh control volume scheme can be used to optimize computation efficiency:

- Conservation equations of mass, energy, and momentum are solved independently;
- A single pressure, p^* , is assumed throughout the system for fluid properties determination. The reference pressure is taken at the top of the core;
- A linear enthalpy profile is assumed for each control volume except for the pool above the core where a uniform enthalpy profile is assumed instead;

- Density is assumed to vary linearly with enthalpy for subcooled liquid and inversely with enthalpy for two-phase mixture; and
- The Homogeneous Equilibrium Model (HEM) for two-phase flow was adopted.

In the MITR flow components, the mass and energy conservation equations for each control volume are, respectively [6]:

$$\frac{dM_i}{dt} = m_{i-1} - m_i \quad (\text{Eqs. C-1})$$

$$\frac{dU_i}{dt} = (mh)_{i-1} - (mh)_i + Q_i \quad (\text{Eqs. C-2})$$

In the above equations, m_{i-1} and m_i represent the mass flow rate entering and leaving control volume i , respectively. The fluid enthalpies at the inlet and exit of control volume i are represented by h_{i-1} and h_i , respectively. The heat source rate of control volume i is represented by Q_i . The total mass and energy content in control volume i , M_i and U_i , are defined by, respectively:

$$M_i = \langle \rho_i \rangle V_i \quad (\text{Eqs. C-3})$$

$$U_i = \langle e_i \rangle V_i - p^* V_i \quad (\text{Eqs. C-4})$$

where $\langle \rho_i \rangle$ is the volume-averaged mixture density, and $\langle e_i \rangle$ is the volume-averaged mixture enthalpy. By assuming a linear enthalpy profile, $\langle \rho_i \rangle$ and $\langle e_i \rangle$ become functions of the inlet and exit enthalpies and pressure:

$$\langle \rho_i \rangle = \frac{1}{h_i - h_{i-1}} \int_{h_{i-1}}^{h_i} \rho(h, p^*) dh \quad (\text{Eqs. C-5})$$

$$\langle e_i \rangle = \frac{1}{h_i - h_{i-1}} \int_{h_{i-1}}^{h_i} \rho(h, p^*) h dh \quad (\text{Eqs. C-6})$$

Applying the approximations that density varies linearly with enthalpy for subcooled liquid and inversely with enthalpy for two-phase mixture, one can solve for the volume-averaged density and enthalpy analytically.

The energy conservation in a finite form is obtained by combining with the mass conservation equation and applying an implicit donor-cell differencing scheme. After applying the conservation equations to the entire system, a system of algebraic equations can be solved for the enthalpy distribution of the system. The system of equations is solved by a Gauss elimination technique. However, in order to solve the simultaneous

equations, the mass flow rates of the system, the core power, and the heat exchangers' energy removal rate must be calculated first.

C.2 Heat Exchanger Model

The heat exchanger energy transfer rate is calculated using the log-mean temperature difference between the primary and secondary system and an overall heat transfer coefficient. The overall heat transfer coefficient is based on the secondary-side heat transfer area and includes the primary and secondary sides convective terms, a conduction term, and a metal heat transfer fouling term. Two types of heat exchangers were modeled: shell-and-tube, and plate type. The heat exchanger metal capacity to store heat was also included in the model.

C.3 Core Power Model

The nuclear power generated in the core comes from two sources: fission and post-fission radioactive decay. A small percentage of this power is deposited in the reflector.

The normalized power produced from fission, P , is calculated using a six-group point kinetics model. In solving the point-kinetics equations, the normalized amplitude function is approximated as linear for each time interval. This assumption permits the delayed neutron precursor equation to be solved analytically and, furthermore, allows for the normalized amplitude equation to be solved numerically using a fully-implicit method.

The normalized decay power, Q_d , produced in the core can be obtained by summing the powers of seven decay-product groups [5]. The group decay constants and fractions were obtained from the ENDF/B-IV calculations based on U^{235} and 10^{13} seconds of irradiation time [31].

The total normalized nuclear power, N , is given by:

$$N = (1 - f_d)P + f_d Q_d \quad (\text{Eqs. C-7})$$

where f_d is the fraction of decay power at steady-state equilibrium conditions.

C.4 Momentum Model

As shown in Figure C-2, The MITR was divided into thirteen control volumes for which conservation of momentum was applied. The control volumes included the core, two regions above the core, the hot leg piping, two pumps, two heat exchangers, the cold leg piping, and four regions of the downcomer.

Figure C-2. Control Volumes of the MITR-III for Momentum Balance

In the MITR system there are two possible flow paths: the "normal" loop in which flow goes through the hot leg piping to the heat exchangers, and the natural circulation loop in which the flow coming out of the core flow shroud re-enters the downcomer through four natural circulation valves. The "normal" loop is used for typical power operations, and the natural circulation loop is used in shutdown mode or possibly low-power testing. These loops are coupled since the flow through the "normal" loop creates a pressure difference across the natural circulation valves maintaining them closed under nominal flow conditions. Several simplifications were made in formulating the momentum governing equations:

- The dynamic head changes in going from one control volume to the next as well as the single-phase acceleration head terms were neglected since they are small compared to friction and form losses;
- The natural circulation valves dynamic behavior was not represented in the model. The valves were assumed to be either fully open or fully closed depending on the pressure difference across the valves;
- The anti-siphon flow path was not modeled except as to prohibit the water level from dropping below the anti-siphon valve height; and
- Small mass flow rates external to the main loop mass flow rate were ignored (e.g., cleanup flow).

The four natural circulation valves were collapsed into an equivalent valve for simplicity. To determine the status of the valve, a force balance on the ball within the

natural circulation valve is applied. The valve status is determined at every time step to determine the correct conservation equations to use in determining the mass flow rates of the system.

To apply conservation of momentum to the "normal" loop it is necessary to determine the head delivered to the fluid by the coolant pumps. The pump head depends on the flow through the pump and on the angular speed of the pump and is determined by solving the pump rotor angular momentum conservation equation along with head-capacity and torque-capacity characteristic curves.

By combining the fluid momentum conservation equations for both possible flow paths along with the pump rotor angular momentum conservation equation, the loop mass flow rates and the pump angular speed can be solved for simultaneously.

C.5 Fuel Heat Transfer Model

An MITR fuel assembly consists of 15 parallel fuel plates separated by coolant channels arranged in a rhombic configuration. Each fuel plate consists of two material regions: a highly enriched Uranium Aluminide fuel meat, clad on both sides by finned Aluminum.

The fuel heat transfer model used is a single-node model developed by J.E. Meyer [32]. In the single-node model, the fuel plate is lumped into one element. The energy conservation equation for the fuel plate is the following:

$$m_p (C_p)_p \frac{dT_p}{dt} = q'' - U_p (T_p - T_b) \quad (\text{Eqs. C-8})$$

where

T_b = bulk coolant temperature

T_p = average plate temperature

U_p = plate overall heat transfer coefficient

q'' = plate surface heat flux assuming a flat cladding (no fins)

m_p = plate mass per unit surface area

$(C_p)_p$ = average heat capacity of the fuel plate

With the heat transfer parameters known, such as the plate overall heat transfer coefficient, the heat conduction equation can be solved numerically using a semi-implicit scheme. The plate overall heat transfer coefficient includes the convective heat transfer coefficient at the surface of the fuel and the conduction through the clad and the fuel. Note that only part of the conduction should be included in the overall heat transfer coefficient since it is based on the temperature difference between the plate average temperature and the bulk coolant temperature and not on the temperature difference between the fuel centerline temperature and the bulk coolant temperature.

Once the average fuel plate temperature is solved for, the average cladding temperature is obtained by using Newton's steady-state law of cooling.

The hot-spot fuel temperatures are calculated in addition to the average fuel temperatures. The same method is utilized except the heat flux, q'' , is multiplied by the radial and axial nuclear peaking factors specified by the user.

The temperature of the fluid exiting the hot-channel is calculated by using equation C-2 with the following modifications:

- The heat source rate, Q , is multiplied by the radial peaking factor specified by the user;
- The flow rate of the hot-channel is multiplied by a flow disparity specified by the user; and
- To account for uncertainties, the heat source rate, Q , is multiplied by the enthalpy-rise engineering hot-channel factor specified by the user.

Given an axial location for the hot-spot, the bulk temperature used in solving the conduction equation is determined by interpolating between the hot-channel inlet and outlet fluid temperatures.

In the laminar flow regime, the Nusselt number is obtained from tabulated data based on a constant axial heat flux profile [33]. In the turbulent flow regime, the heat transfer coefficient was calculated by the correlation developed by Seider and Tate. The hydraulic diameter used in determining the Reynolds number is based on a flat cladding approximation (maintaining the same coolant channel area and clad mass). The heat transfer rate improvement provided by the existence of fins in the clad is accounted by multiplying the heat transfer coefficient by a user-specified fin surface effectiveness. To account for uncertainties, the heat transfer coefficient is divided by the heat-flux engineering hot-channel factor specified by the user.

C.6 Onset of Nucleate Boiling

The Bergles/Roshenow onset of nucleate boiling correlation is used to determine the heat flux necessary to create the onset of nucleate boiling. The incipient heat flux determined by this correlation is relatively low, as this relation was based on visual observation of the first nucleation occurrence rather than on observation of wall temperature response, which requires a significant number of nucleation bubbles. Therefore, the onset of nucleate boiling is calculated conservatively.

C.7 Future Recommended Work

A more detailed core analysis method can be incorporated with the TAMLOC system analysis to make detailed core transient studies. Initial work on the detailed core analysis is described by McGuire in the next appendix (Appendix D).

Appendix D: MITR Multi-Channel Core Analysis

D.1 Pressure Drop Instability in MITR

The current MITR-II core contains 27 rhombic assembly locations making up a hexagonal core. These 27 locations can be filled with a combination of fuel assemblies, unfueled (dummy) assemblies, and experiments. These locations all receive coolant from a common lower plenum and discharge the heated coolant to a common outlet plenum. In addition to these coolant flow paths, there are possible diversion flows that would bypass the heated core.

This configuration presents the possibility for a pressure drop (Ledinegg) instability that could lead to a flow excursion. If this flow excursion could cause premature CHF then the instability should be regarded as a limiting condition. Previous analyses of MITR-II have indicated that flow instability-induced CHF will occur prior to normal CHF. Because channel voiding is a precursor to this type of flow instability, the MITR-II Limiting Safety System Setting is at the Onset of Nucleate Boiling (ONB).

The prediction of the Onset of Flow Instability (OFI) is important for any accident analysis that includes the appearance of two-phase flow anywhere in the core. Due to radial power gradients and inlet flow disparities, there is a distribution of outlet enthalpies from the flow channels (i.e., each of the 27 assembly positions plus any diversion paths). When the hottest channel first begins to boil, it will have a higher pressure drop than the single-phase channels due to the rise in the acceleration and friction pressure drop components. All channels will have the same pressure drop because they share a common set of plena. Therefore, the boiling channel will have to receive a lower mass flow rate in order to have the same pressure drop as the two-phase channels. The lower flow rate will cause more voiding, leading to even less flow. This excursive behavior will continue until the gravity term overcomes the other pressure drop terms and stabilizes the flow rate.

D.2 Past Work

The formation of voids in a channel is a necessary but not a sufficient condition for flow instability. The LSSS is conservatively set at the point where the very first vapor void forms in the core (ONB). Using this limit, a single lumped-channel model of the core with appropriate peaking and disparity factors can be used to model the behavior of the core up to the LSSS. This analysis has been done by Trosman and has yielded a set of proposed operating and LSSS points based on a series of transients at 5 and 10 MW.

In order to analyze the thermal-hydraulic performance of the core during accidents which may lead to the onset of two-phase flow in a channel it is necessary to accurately predict the Onset of Nucleate Boiling (ONB), the Onset of Significant Voids (OSV), and the Onset of Flow Instability (OFI). A two-phase, single channel code could be used to predict ONB and OSV using existing semi-empirical correlations. Such a code would need to determine the enthalpy, pressure, and flow rate at each axial node by solving a set

of energy, momentum, and mass conservation equations. Using this information it is possible to estimate the OFI by plotting the pressure drop versus flow rate (see Figure D-1). For most systems the location of the local minimum in the curve is a good estimate of the OFI. Flow Instability experiments have been done with single channels to determine the points needed to make a plot similar to Figure D-1. These experiments have yielded many correlations for OFI under various conditions. Strictly speaking, the minimum in the pressure drop-flow rate curve is not OFI, but it is the point at which you would expect to see a flow excursion begin if there was an alternate flow path of lower resistance available for the fluid.

The single channel codes and experiments are useful tools for estimating the conditions necessary for OFI, but they have several shortcomings:

- 1) They are generated using a series of steady-state measurements, whereas in a real application the system will probably not reach a steady state (equilibrium).
- 2) They cannot predict the resulting flow distribution among the channels after OFI is reached in one or more channels.
- 3) They cannot determine whether the flow excursion resulting from OFI will be sufficient to cause premature CHF.
- 4) They do not include the effects of the rest of the coolant loop that is found in a real system (pump head, heat exchangers, etc.).

If the objective of an analysis is to predict the reaction of a multichannel system to a transient that could cause OFI, it is necessary to model all possible parallel flow paths separately.

D.3 Multi-Channel Core Analysis Code For MITR

In order to analyze the behavior of the system under two-phase conditions a multi-channel core analysis program is being developed for MITR. The code is applicable to either the 5 MW or 10 MW case because the core configuration will remain the same, and the code allows component characteristics (heat exchangers, pumps, etc.) to be altered. The following is a summary of the work in progress on the computer program development.

D.3.1 Conservation Equations

Only a brief description of the conservation equations used to model MITR is presented here because they are similar to the governing equations used in most thermal-hydraulic codes. The energy equation is nodalized and set up to use a donor-cell finite

Figure D-1. Pressure Drop Versus Flow Rate Curves Showing the Onset of Flow Instability (OFI) for MITR at 5 MW and 10 MW

difference solution scheme. At each time step, enthalpy is "transported" along the channel from the donor node to the recipient node. In the recipient node, the heat flux that the fluid passed through is added and a new set of properties (x , α , h , ρ) is determined. A fixed recipient control volume and a variable donor node volume are used to allow time steps greater than the characteristic node transport time ($\Delta z/v$) to be used.

The momentum equation balances the pressure drops and the pump head to determine the change in loop flow rate at each time step. This loop flow rate is split among the parallel channels under the constraint that all pressure drops must be equal. The code is set up to use either a Momentum Integral or Single Velocity model. A single mass conservation equation is used so there are no exchange or interface terms. The phasic distribution is redetermined at each axial node for each time step. The flow split among channels is solved implicitly with the enthalpy transport equations so that the void distribution converges at each time step.

D.3.2 Key Constitutive Relations

The accurate determination of OFI is based on an ability to locate OSV and to determine the pressure drop during two-phase flow. The models described below have been chosen to optimize the code for the MITR conditions.

Onset of Nucleate Boiling

The onset of nucleate boiling is important in OFI determination because it is the point where the void fraction becomes greater than zero. This adds an acceleration pressure drop and a two-phase friction multiplier to the pressure drop equation, and it increases the single-phase friction factor by adding surface roughness to the pipe.

These effects are expected to have a small effect on the location of OFI and a negligible effect on the post-OFI flow disparities. The current model assumes that the void fraction is zero until OSV. The ONB effect will be added later using the correlations of Davis and Anderson [17] or Cubillos-Moreno, et. al. [34] which are applicable near atmospheric pressure.

Onset of Significant Void

The OSV model of Saha and Zuber used in many thermal-hydraulic codes is inadequate to describe OSV in the MITR due to its finned fuel surfaces [27]. Kowalski performed a series of experiments on finned fuel plates to determine an applicable correlation for OSV [35]. He used the same dimensionless coordinates as Saha and Zuber, and found the following correlation to be valid under the conditions found in MITR:

$$St = 0.0446Pe^{-0.18} \quad (\text{Eqs. D-1})$$

This result implies a dependence on flow rate even at high Peclet numbers (Saha and Zuber assumed that flow dependence disappears above $Pe = 70,000$). Rogers et. al. derived a semi-empirical method for predicting OSV based on a force balance on a growing bubble. When the buoyancy and bubble drag forces are sufficient to overcome the surface tension, the bubble detaches and OSV is reached. Their correlation is the most appropriate for low-velocity, low-pressure systems (such as MITR), but its implicit relationships make it difficult and costly to use in a code. However, Rogers presents a relationship for predicting the vapor fraction at OSV that will be used in the code [36]:

$$\alpha_{OSV} = \frac{1.05 C_1(r_b)}{D_H} \quad (\text{Eqs. D-2})$$

Where,

r_b = Bubble diameter at departure

D_H = Hydraulic diameter

C_1 = A weak function of wetting angle

Void Fraction Relations

In the current version of the code the void fraction is assumed to be zero until OSV. A linear relationship will be assumed between ONB and OSV, up to α_{OSV} found from equation D-2. Beyond OSV the void fraction continues to rise rapidly in low pressure systems.

With void fraction and equilibrium quality both known at OSV (from Equations D-1 and D-2), the flowing quality can be found by a curve fit similar to that developed by Levy [37]:

$$x(z) = x_e(z) - x_e(z_{ONB}) \exp[(x_e(z)/x_e(z_{ONB})) - 1] \quad (\text{Eqs. D-3})$$

Where,

$x(z)$ = Flowing quality at z

$x_e(z)$ = Equilibrium quality at z

$x_e(z_{ONB})$ = Equilibrium quality at ONB

Once the flowing quality is known along the length of the channel, the drift flux model is used to find the void fraction. The drift flux model can be expressed in terms of two semi-empirical parameters. A distribution parameter, C_0 , accounts for the void and velocity profile effects. The other parameter is the slip velocity (u_{vj}). These parameters depend on the flow regime, which can be calculated by using a flow map appropriate to the system conditions. This method requires that efforts be made to smooth out discontinuities that result from flow regime transitions. The code described here accomplishes this by using weighted values of C_0 and u_{vj} in the transition regions between flow regimes.

Another approach is to use a continuous (flow regime-independent) void correlation such as the Chexal-Lellouche relationship [38]. This has been shown to be valid over a wide range of conditions, but it is cumbersome to apply.

Two-Phase Friction Multiplier

The two-phase friction multipliers included in the code are the Homogeneous Equilibrium Model (HEM) and the Martinelli-Nelson Model. For mass fluxes above 2000 kg/m²s, the Martinelli-Nelson model is found to be the most accurate, while the HEM model gives better predictions at low flow rates.

D.3.3 Input Parameters

The location of OFI and the post-OFI flow disparities in the core are complex functions of the interactions between all of the channels that share the common plena. The effect of some of these parameters are described below.

Heat Flux Shape

A single-channel model to predict the location of OFI usually simulates the hottest channel because this is where the instability is expected to occur first. This is not always a valid assumption, and it should be verified before being relied upon to predict safety limits.

To model the array of parallel channels some knowledge of the heat flux shape must be available. Normally this is similar to the power shape which can be obtained from neutronic analyses or core power monitors. The difference between the thermal and neutronic power shapes is the result of (1) a time delay between the neutronic power and the heat flux due to the transport of energy through the fuel and clad; and (2) heat losses due to fuel assembly conduction and gamma energy deposition outside of the core. The radial shape will determine which channel receives the greatest amount of absolute heat addition. This is not necessarily the channel that has the highest enthalpy rise (or highest outlet temperature). It is the radial peaking factor that is responsible for the shape of the pressure drop vs. mass flow curves.

The axial peaking factors describe where along a given channel the heat is added. In a study at Columbia University various axial power shapes were used to generate the same total channel power [39]. The flow rate corresponding to OFI was found to be slightly lower if a non-uniform power shape was used. This result was found regardless of how the power shape was skewed (double peaked, cosine, etc.). This suggests that there is a contribution of local conditions to OFI that is not accounted for in the semi-empirical models of Whittle and Forgan and others. This would indicate that performing an analysis with a uniform axial shape would give a conservative result. However, the Columbia study did not include a power shape that peaks sharply at the top. A top-peaked axial shape should be studied because (1) the pressure gradient components which trigger OFI peak at the exit; and (2) pool-type research reactors have an axial power peak at the top

due to the water above the core. The MITR multi-channel code uses local heat flux conditions to calculate OSV.

The axial and radial powers at each time step are currently provided to the code as input parameters. After the thermal-hydraulic model is worked out it will be coupled with a space kinetics code to provide power shape information and to receive neutronic feedback from the coolant. The triangular mesh code developed by DeLorey appears to be the best choice for a neutronic model [40].

Plenum Flow Disparity

The objective of an OFI study to predict safety limits is to find the flow in each channel at each time step. The flow split in a system of heated channels is a function of both the channel pressure drops and the inlet plenum pressure gradient. Even at zero power a flow disparity exists in the core due to the shape of the plenum. When the reactor is at power, the flow distribution will be a result of the effects of the plenum disparity and the radial power shape.

In order to model the lower plenum pressure gradient it is assumed that the pressure drop from the plenum plate to the core outlet is the same for each channel. Based on a set of plenum flow disparities measured at zero power, a plenum disparity vector can be solved for. This vector, when treated as a collection of form loss coefficients, will reconstruct the desired plenum disparity in the code. Using this approach the plenum disparity effects decrease as flow rate is dropped, as would be expected intuitively.

Channel Geometry

The channel geometry influences the OFI and post-OFI behavior of the system in four ways. First, any differences between channels will lead to different pressure drop characteristics (e.g., a smaller hydraulic diameter will give larger friction losses). A prediction of OFI must therefore allow the user to input different geometries for each channel to investigate the effects of manufacturing tolerances or flow obstructions. The shape of the channel has been ignored in most correlations for OSV because equivalent diameters are used to correlate the data. Sardy reviewed the data from several OFI studies and used statistical analyses to distinguish between the correlations for rectangular and cylindrical channels [41] (see Appendix J). His least-square fits can be expressed in forms similar to Kowalski:

$$St = 0.0137Pe^{-0.024} \quad \text{For Rectangular Channels} \quad (\text{Eqs D-4})$$

$$St = 0.1076Pe^{-0.219} \quad \text{For Circular Channels} \quad (\text{Eqs. D-5})$$

Secondly, the length and diameter of the channel affect the friction losses. A long two-phase length increases the frictional pressure drop, which is a destabilizing component.

The diameter also plays a role in the flow instability phenomenon in plate-type research reactors with narrow coolant channels like the MITR. If the distance between coolant channels at closest approach is on the order of bubble diameters at departure, OFI may occur before bubble detachment because the bubbles can contact each other during growth. Parra estimated that this will not occur under the MITR conditions [17].

Finally, the presence of fins on the fuel plates can change the character of the system. The fins cause a nonuniform heat flux along the walls of the channel. Parra reported that the heat flux at the fin tip is 44 % higher than that in between fins [17]. The local heat flux determines the point of ONB and the clad temperatures. The presence of fins also presents a question of what value to use for the hydraulic diameter and equivalent (heated) diameter. The answer is the hydraulic and equivalent diameters which give the correct pressure gradients and heat transfer coefficients should be used. In other words, experimental results or detailed Computational Fluid Dynamics (CFD) models are required to verify the choices for these diameters.

Other Parameters

The inlet subcooling obviously affects the stability of the channel because it determines the exit quality. The entrance and exit effects must be included in any model of the complete system. The entrance effects can serve as a stabilizing influence in a potentially unstable system [34]. In addition, the rest of the loop must be described in detail in order to compute the total system pressure drop. The addition of the rest of the loop in the calculation has a stabilizing effect. Whittle and Forgan state that analysis of a single heated channel cannot be used to predict OFI when the out-of-core pressure drop is large compared to the core [42].

D.4 Application of Multi-Channel Code

The MITR multi-channel model will be used to predict the behavior of the system under a variety of transient and accident conditions. These runs will be used to:

- 1) Accurately determine when and where OFI will occur in the core under given conditions.
- 2) Estimate the severity of the flow excursion. This will provide flow rates as a function of time in each channel during the transient.
- 3) Indicate whether a premature (i.e., OFI-induced) CHF occurred in one of the channels due to dryout.
- 4) Predict the maximum fuel temperature in the core up to the point of CHF (either normal or premature).

Transients and accidents such as loss of flow, loss of heat sink, loss of coolant, flow blockage, and anticipated transient without scram (ATWS) can be analyzed under different

initial conditions to verify or revise the LSSS and operating limits proposed by Trosman [6].

E.1 Introduction

In order to operate MITR at higher powers, it is necessary to have increased uranium loadings in the fuel. These loadings will also provide the benefit of operating the reactor with more reasonable operating lengths between refuelings. Extended Life Aluminide Fuel (ELAF) makes this option available for the reactor. The ELAF fuel assemblies are structurally the same as those in the current MITR-II core with the major difference being higher U_{ALx} densities in the fuel meat. Hence, higher fissile isotope concentrations will be in each fuel assembly. This higher uranium loading will cause an increase in the reactivity of the core, which can be held down through the insertion of a burnable poison into the fuel plates. Because of its success in previous ELAF studies and irradiation tests, B_4C will be chosen as the burnable poison. Burnup of this ELAF fuel will be limited by the peak fission density point, chosen as $3-4 \times 10^{21}$ fissions/cm³ (almost twice the current MITR value). This limit was achieved at about 50 % burnup of the fuel plates in the University of Missouri research reactor (MURR) [44]. Two interesting issues of the ELAF fuel will be studied: the uranium loading which will satisfy the fission density limit while maximizing burnup, and the necessary burnable poison concentration to keep the reactivity of the core held down.

Although the CITATION code system for reactor analysis [45] has proved to be a useful tool in the past, better results can be obtained with a newer code system. The code system proposed for use in this analysis is the ORNL code system of SCALE/BOLD VENTURE/VALE. This code system runs on numerous platforms, but the majority of the work will be executed on a PC system. SCALE is a modular code system for performing criticality safety analyses. It also functions as part of the system that will allow for spectrum weighting of cross sections for analysis (through its two main modules, NITAWL-II and XSDRN-PM). BOLD VENTURE is a multidimensional, multigroup diffusion code that includes numerous modules that allow for depletion and fuel management [46]. VALE is a stand-alone code that acts as an additional module to BOLD VENTURE. It is the triangular neutronics module of the code system [47]. Although the code system is not an easy one to manipulate, it has been shown to produce extremely good results [48].

E.2 Calculation Process

In order to obtain reliable results with any neutronics code, it is necessary to have accurate cross sections as input. The SCALE system and the scheme we have derived for its use provide an accurate tool for cross section generation. A diagram of this cross section generation scheme appears in Figure E-1. Microscopic cross sections are referenced from a master 27 group cross section library (based on ENDF/B-IV). Resonance calculations are performed on the fuel meat through the Nordheim Integral resonance treatment method found in the SCALE module NITAWL. Through transport theory flux calculations, XSDRN-PM (another module of SCALE) then provides accurate unit cell spectra weighting of the cross sections. A detailed one dimensional cut of the

core is then input into XRDRN-PM to provide spectrum weighting of the cross sections for the core. In this detailed 1D-cut; the fuel, clad, water, and core structure material are explicitly represented as best as possible to reproduce a detailed cut across the reactor. The cross sections are then region weighted in each of the rings of the core (A,B,C). This region weighting is necessary because of the extremely different spectra, caused by the small core size and large reflector surrounding it. This procedure produces a microscopic cross section set that helps to increase the accuracy of the neutronics code through reliable microscopic cross sections.

Figure E-1. Cross Section Generation Scheme

The next step of the process will be to homogenize the MITR core into triangles which will allow for ease of use in fuel management, and not compromise accuracy. A sample of the core triangularization is shown in Figure E-2. This division of the core allows for two important features. The first is that assemblies are almost explicitly represented, yielding simple fuel management manipulation. Other triangularization methods produced arrangements in which each section of the core had triangles that were not core symmetric, as this one is. The other important characteristic of this triangularization is that the control blade triangles only consist of the blades and core support housing, an important aspect to not adversely effect fuel depletion results. Dividing the core with this arrangement will yield a final triangular-z model of the core with approximately 52,000 triangles (or mesh regions) being well within the capabilities of the code and computer system.

Once the number densities for triangular-z model are created and combined with the microscopic cross section set earlier produced, a neutronics model of the core can be created. As stated earlier the neutronics calculation will be performed in the ORNL code

Figure E-2. Core Triangularization Scheme

of VALE. The BOLD VENTURE system will allow fuel management process such as depletion to be performed on the reactor model.

E.3 Verification of the Code System

Upon introduction of this code system and use, two major issues were discussed as to whether or not the code will produce reasonable and reliable results. The first of these was the neglecting of the side plates in the cross section generation processes, which is caused by the one-dimensional nature of the SCALE system. The other issue concerned the ability of the triangular diffusion code to yield accurate data. It was decided to run a test problem through the system to check its abilities. This results of the test problem would then be compared to an exact representation of the problem in the MCNP program. The test problem was a 2x2 assembly array as shown in Figure E-3. The MITR assemblies were surrounded by a water and heavy water reflector. In the testing of the system the assemblies were smeared, while in the MCNP calculation they were exactly represented.

Results for the comparison of the two methods are shown in Table E-1. Two different tests of the SCALE-VALE system were performed. The difference between the two was the method of the 1D-cut for the spectra weighting. The first case had an explicit representation of the assemblies for the 1D-cut across the water, assemblies and then heavy water. In contrast, the second case was with the assemblies homogenized in the 1D-cut. The table shows that the use of either method produced reliable results. As expected, when the assemblies are homogenized, an increase in k_{eff} occurs, due to a loss in absorption in localized regions near the fuel plates. Performing the detailed cut does not lead to a tremendous increase in CPU time or setup time, so this method will be used. It was concluded that reliable results can be obtained from this code system.

Table E-1
Results of Test Case

RUN TYPE	K-eff	% error (from MCNP)
MCNP	0.76335	N/A
SCALE/VALE (detailed 1D-cross section weighting)	0.769627	0.82%
SCALE/VALE (homogenized region weighting)	0.788237	3.26%

- 1 - H₂O
- 2- Smeared Fuel Assembly
- 3 - D₂O

Figure E-3. Diagram of Test Problem (Each triangle leg is 7 cm long)

E.4 Current and Future Work

At the time of the writing of this report, a system for calculation of the number densities for the core model is being developed in FORTRAN. Essentially this program will allow the user to input the assembly loadings, and the triangular core model will be the output. After this code is completed, actual work on the MITR core will be performed. This will include determination of the correct ELAF fuel and burnable poison loading, and suggestions for some fuel management schemes. It is hoped that this code system will be a useful tool for current and future MITR core design work, and prove that the ELAF fuel can provide the necessary requirements for higher power operation.

F.1 Introduction

The purpose of this investigation was to study the use of PARET code for determining the step reactivity insertion that would produce a clad temperature of 450°C for the MITR. The limit of 450°C was chosen due to the softening of aluminum at this temperature.

The present reactivity limit in the SAR was determined by using correlations drawn from experimental results of the SPERT [50] cores. It is desirable to use a modern transient analysis code such as PARET to update this reactivity limit. The following sections present the results obtained in this study.

F.2 Results

First, the worst initial conditions for the core were determined. It was found that for large reactivity insertions (i.e., above 2 dollars) the lower the initial power of the core, the higher the peak clad temperature. Also, the channel modeled with a power distribution peaking at the bottom of the core gave the most limiting results.

By investigating the size of the time-step increment to use, large numerical instabilities in PARET were discovered. These instabilities were discovered by noticing that the calculated outlet flow behaved randomly. Therefore, a detailed study of the time step increment size was performed. Two converging areas were observed as seen in Figure F-1. For all subsequent calculations, a time-step size corresponding to the smallest of the two converging areas was used. This would obviously result in higher peak clad temperatures than if a larger time-step size was used. However, since this time step size was small (approximately 5×10^{-7} seconds), there was concern about the possibility of round-off errors (i.e., chopping errors). This was investigated further and it was determined that round-off errors would not occur as long as values larger than 5×10^{-7} seconds is used.

The fuel was first modeled as a fuel "meat" clad by flat aluminum plates. However, the cladding has fins which increase the heat transfer area and also increase the pressure losses. Therefore, to take into account the fins, the fuel was modeled as shown in Figure F-2. The fuel plate and associated coolant channel width was doubled while the coolant channel thickness was halved to keep the same amount of coolant. In addition, the fuel and clad specific heat were also halved to be able to accurately represent the amount of stored heat during the transients studied. The results indicate that lower peak clad temperatures are obtained when the fins are included in the model.

Figure F-1. Clad Temperature Peak Versus Time Step Increment Size

Figure F-2. Modeling of the fins

The following table summarizes all of the different limits determined in this study. Most were calculated for a step reactivity insertion while some reactivity ramps were studied as well.

Table F-1
Reactivity Insertion Limits

Core Initial Conditions	Average Channel	Hot Channel	Hot Channel (0.1 s ramp)	Hot Channel (0.5 s ramp)	Hot Channel (1 s ramp)	Hot Channel (with fins)
5 MW 2000 gpm	1.5\$-1.6\$	1.34\$	N/A	N/A	N/A	1.39\$
5 W 2000 gpm	N/A	1.45\$	N/A	N/A	N/A	N/A
5 W 2 kg/s	N/A	1.3\$-1.33\$	1.2\$-1.3\$	1.2\$-1.27\$	1.2\$-1.3\$	1.4\$

It should be noted that there is a question of the instability noted as the time step size was changed and the validation of the PARET code at small time step sizes. Further studies with other transient analyses codes should be made to complete the investigation of this reactivity insertion limit.

Appendix G: Radiation Shielding Study [51]

G.1 Introduction

Shielding implies the deliberate introduction of material between a radiation source and an object(s) to reduce the radiation intensity and hence damage to the object. The object(s) of most concern are personnel; operators, faculty, students or visitors who come within the vicinity of the MITR. Pursuant to 10CFR20 Subpart B (Radiation Protection Programs), paragraph 20.1101, the responsibility of radiation protection for the MITR is managed and administered by the Reactor Radiation Protection Office (RRPO).

Surrounding the MITR graphite reflector is the thermal shield which absorbs most of the gamma energy from the core gamma rays. The thermal shield consists of a radial cylindrical section, a bottom section, and a thermal column section. The radial section consists of two cylinders of 2 inch steel each surrounding an annulus of 1 1/2 inches of lead. Imbedded within the lead are stainless steel cooling coils for cooling the entire (radial, bottom, and thermal column) thermal shield. Approximately 1.3 % of the operating reactor power is removed by the thermal shield cooling system. Openings are provided for the experimental beam ports and the thermal column region. All internal surfaces which are exposed to a neutron flux are lined with 0.25 inches of Boral. The openings for the beam ports are lined with cadmium and have cadmium collars which overlap the Boral. The bottom section of the thermal shield connects to the radial cylindrical thermal shield and is similarly constructed. Openings in the bottom section are for the heavy water reflector piping and the basement medical facility shutter tank. The thermal column portion of the thermal shield has a frame with 1/16 inch aluminum sheeting attached. This is to form a gas tight seal between the graphite and the thermal column. A cadmium shutter is also provided to control experimental use of the thermal column and hohlraum region within the Blanket Test Facility (BTF). Attached to the cadmium shutter frame is a CO₂ filled aluminum box to displace air and thus reduce the production of Argon-41. The entire thermal column is shielded with a 1 inch steel shell lined with 1/4 inch of Boral and then backed with 3 inches of lead.

A biological shield surrounds the thermal shield on all sides of the reactor. It consists of heavy or dense concrete made of Portland cement, magnetite iron fines, and steel punchings with an average density of 280 lbs/ft³. The reactor and radial reflector is surrounded by the lower and upper annular rings, the upper shield annular ring, the upper shield access ring and the top shield lid. A new reactor top shield lid was designed and installed in May 1993 and is slightly different in construction to the previous lid described in the Reactor Systems Manual. Surrounding these rings is the side biological shield, consisting of approximately five and a half feet of high density concrete. The reactor faces are then covered with 3/8 inch steel plating. The shielding was designed for a dose rate of less than 1 mrem/hr at the top and sides when the reactor is operating at 5 MW. The bottom biological shield contains 50 1/2 inches of high density concrete between the bottom thermal shield and the ceiling of the basement medical therapy room.

Experimental horizontal ports penetrate the biological shield to provide neutron beams for irradiation facilities, experiments and nuclear instruments. The ports consist of one 12 inch radial port (12SH1), four 6 inch radial ports (6SH1-4), six 4 inch radial ports (4DH1-6), two 6 inch semi-radial ports (6RH1-2), one 6 inch through port (6TH1/2), two 4 inch through ports (4TH1/2, 4TH3/4), and four 4 inch tangential ports (4IH1-4). All consist of a port box, a stainless steel port sleeve, an aluminum port liner, and a shield plug.

G.2 Results

G.2.1 Results from Reactor Start Up

The gamma and neutron radiation levels external to the biological shield are directly related to the reactor power level. During the fission process, core fission gamma rays are also emitted. Subsequent to this, a portion of the fast neutron flux is thermalized and neutrons are captured producing capture gammas. Also, there are small radiation contributors such as aluminum and hydrogen activation. In total however, the neutron and gamma radiation level external to the biological shield should be proportional to the reactor power level.

The gamma and neutron radiation levels at an undisturbed section of biological shielding were measured on a typical reactor startup. The areas chosen were in the vicinity of the 6HS3 port on the reactor floor and above this location on the platform above 4DH4. These locations are at the same height as the bottom and top of the reactor core respectively. The results are shown below in Table G-1. The results below also indicate the 5 MW and 10 MW extrapolated values since data could only be obtained up to 4.5 MW.

Table G-1
Radiation Levels in the Vicinity of Port 6HS3 and Platform

Vicinity of 6HS3 Port			Vicinity of Platform (Above 4DH4)	
Power (MW)	Gamma (mr/hr)	Neutron (mrem/hr)	Gamma (mr/hr)	Neutron (mrem/hr)
1	0.4	0.05	0.4	0.1
2	0.55	0.1	0.6	0.2
3	0.7	0.13	0.7	0.3
4	0.85	0.15	0.9	0.45
4.5	1	0.2	1	0.5
*5	1.1	0.22	1.1	0.56
*10	2	0.44	2	1.11

* Denotes extrapolated values.

G.2.2 Results at Full Power

Numerous gamma and neutron radiation measurements were taken on several separate occasions. Below is a summary of the most notable locations with the 5 MW and 10 MW extrapolated values shown.

Table G-2
Radiation Levels at Various Locations

Location	Gamma (mr/hr)*	Neutron (mrem/hr)*	Gamma (mr/hr)**	Neutron (mrem/hr)**
Blanket Test Facility (cadmium shutter closed)	0.13-0.25	0.13	0.26-0.5	0.26
Vicinity 4DH3	2.22	1.1	4.44	2.2
ER Door (spent fuel side)	1-1.3	NDN	4.44-10	N/A
Reactor Top Deck	12-110	13-83	26-220	26-166
Reactor Top (general area)	1-15	6	2-30	12

* Denotes extrapolated value for 5 MW

** Denotes extrapolated value for 10 MW

NDN denotes no detectable neutrons

G.2.3 Results from Reactor Radiation Protection Office (RRPO)

The reported study also contains detailed results of the RRPO gamma and neutron surveys from monthly and annual radiation surveys over the last few years. The results are all extrapolated to 5 MW and were obtained for every non shutdown periodic survey. The data is useful obtaining a range of expected gamma and radiation levels for 5 MW. The fast and slow neutron radiation results are not easily related, since the neutron detector used for this study accounts for both fast and slow neutrons in indicating a biological dose. However, the gamma radiation results are directly related since the detectors used were similar. It is recognized that there are other factors for the variation in data, such as: human error, instrument responsiveness, background radiation, etc.; however, the general trend is very useful in predicting the expected radiation levels. The RRPO survey locations chosen to report most closely replicated the locations data was taken for this study.

G.2.4 Shield Photographs

To document the current state of the installed shielding, black and white photographs were taken. It is hoped that these photographs can be utilized for gaining a knowledge of the biological shielding without actually entering into a radiation area.

G.3 Conclusions

The results of this study indicate that with the current shielding at 5 MW the dose rates in some areas are greater than 1 mrem/hr. The 1 mrem/hr threshold is referenced in the Reactor Systems Manual and the Safety Analysis Report for MITR-II. This threshold

is very conservative and all exposures are kept well below the established limits (e.g., 10CFR20). The wording in the Reactor Systems Manual and the Safety Analysis Report for the proposed MITR-III should be modified to explain the 1 mrem/hr basis with regard to the solid portions of the shield design at 5 MW. Although the present shielding could be used at operating levels up to 10 MW, consideration might be given to added protection in the following areas:

- The equipment room doors in the basement (from N^{16} gammas).
- The upper corner of the chemistry room where there is free communication with the equipment room (N^{16} gammas).
- The experimental horizontal ports where the shielding can be improved with a more uniform concrete and/or polyethylene slab material. An interlocking design could minimize the scattering paths of the present shielding.
- The seams in the BTF. The BTF is a strong source of gamma radiation due to the activation of Ar^{41} . The seams should be filled with a pliable putty beta/gamma shielding material.
- The reactor top area shielding could be increased, especially to support 10 MW operations. Additionally or in conjunction with increased shielding, the top area can be administratively controlled differently than it is presently. A high radiation area will result on the top with 10 MW operations. Permanent posting of this area or automatic high radiation or very high radiation area controls could also be instituted.

Appendix H: Preliminary Evaluation of Radioactive Effluents for an Upgraded MITR-III [52]

H.1 Objectives

The objective of this study is to attempt to answer the following questions:

1. What are the impacts of the new Part 20 (10CFR20) on the current operations of MITR-II?
2. What are the impacts of the new Part 20 on the redesign of a potential 10 MW MITR-III reactor? (included should be considerations of future trends).
3. Are there linear relationships in the effluents between 5 and 10 MW operations?
4. How can compliance with future limits be ensured (how can problems be mitigated)?

H.2 Production and Release of Radionuclides

Radionuclides are produced in the operation of the MITR. Routine controlled releases of radionuclides from MITR occur regularly through two pathways: liquid wastes into the municipal sanitary sewer system and gaseous releases through the reactor's stack. This study focus exclusively on the releases through the stack. Figure H-1 shows the various sources of air coming from the facility and the pathways to the stack.

Figure H-1. Schematic Diagram of the MITR-II Ventilation System

The MIT reactor emits a number of radionuclides into the environment in small amounts. These effluents and their relationship to NRC release limits are shown in the following table.

Table H-1
Effluent Releases Relative to NRC Regulations
(Based Upon Fiscal Year 1993 Stack Effluents)

Effluent	% of MPC ¹	% of DAC ²	% of DAC at 10 MW ³ (projected)
Kr ^{85m}	0.101	0.101	0.202
Kr ⁸⁷	2.79	2.79	5.58
Kr ⁸⁸	2.13	4.74	9.48
Xe ¹³⁵	0.433	0.62	1.24
Xe ^{135m}	0.529	0.40	0.80
Xe ¹³⁸	2.35	3.52	7.04
Xe ¹³³	0.016	0.01	0.02
I ¹³¹	0.285	0.14	0.28
I ¹³³	8.15e-5	3.3e-5	6.6e-5
I ¹³⁵	1.60e-5	1.6e-6	3.2e-6
I ¹³⁴	3.32e-4	5.5e-4	1.1e-3
I ¹³²	3.80e-5	5.7e-6	1.1e-5
Ar ⁴¹	6.0	24.0	48.0
H ³	0.004	0.008	0.016
TOTAL	14.6	36.3	72.7

¹ The NRC standard used prior to 1/1/94 revision of 10CFR20.

² The NRC standard used effective 1/1/94 due to revision of 10CFR20.

³ This assumes a linear relationship between 5 and 10 MW.

Argon-41

As can be seen in Table H-1, Argon-41 is the most significant radionuclide released at the MIT Research Reactor. The Ar⁴¹ is produced when air becomes irradiated and Ar⁴⁰ undergoes a neutron capture reaction. Air typically consists of 0.93 % Ar⁴⁰, which has a neutron activation cross section of 0.65 barns [53]. Since the activation depends on the neutron flux, a linear relationship between Argon-41 activity and power can be assumed.

In the mid-1980's two MIT Masters Theses (Reilly, 1984; Deplitch, 1985) focused on the reduction of Ar⁴¹ emissions. These theses not only produced some positive results in reducing the emissions, but also developed some valuable conclusions:

1. Better sealing of ports will reduce the air flow reducing the Argon-41 production
2. The lead shutter should be studied in order to find out its contributions to the Argon-41 production.

3. The thermal column also appears to contribute to the Argon-41 production, better sealing of this large complex structure will be pursued.
4. Temperature changes in the reactor building can cause drafts which affect the in-flux spaces changing the production of Argon-41.

Fission Products

The MITR also emits small amounts of fission products. The emissions come from two sources: defective fuel and impurities on the surfaces of the fuel. Assuming that there are no major changes in the fuel (the fuel remaining the same and burned twice as fast), the production of fission products from impurities should double. Although the effect of doubling the flux on the cladding defects has not been investigated, a linear assumption is taken between the production of fission products effluents and the power.

H.3 Regulations on Effluents

NRC Regulations

The NRC standards are based upon maximum concentrations of released effluents. Effective January 1, 1994, 10CFR20 - *Standards for Protection Against Radiation* - which establishes the standards was amended. In the amendment, the emission standard was changed to the Derived Air Concentration (DAC) from the Maximum Permissible Concentration (MPC). In general, the new DAC limits are approximately 50 % of the MPC limits. The new standards state that "The total effective dose equivalent to individual members of the public from the licensed operation does not exceed 0.1 rem (1 mSv) in a year, exclusive of the dose contribution from the licensee's disposal of radioactive material into sanitary sewage (10CFR20, page 38)."

One confusing aspect of this change is that although the basis was reduced by 80 %, the drop in allowed concentration was only by about 50 %. This was caused by the switch from a Maximum Permissible Concentration (which assumes a 24 hour, 365 days per year exposure) to the Derived Air Concentration standard (which assumes a 2000 hour per year exposure). There is another complicating factor. The "Effluent Concentration" limits for individual radionuclides, published in Appendix B of Part 20, "are equivalent to the radionuclide concentrations which, if inhaled or ingested continually over the course of a year would produce a total effective dose equivalent of 50 mrem (10CFR20, page 51)." Thus, the limits which give the predicted MITR-III an exposure rate of 70-75 % of the total allowable effluence are doubly conservative.

EPA Standards

While the NRC uses maximum concentrations for its release limits, the EPA uses dose-based standards. These are promulgated in 10CFR61 - *National Emission Standards for Hazardous Air Pollutants (NESHAPS)*. Under the NESHAPS regulations, facilities

are limited to a dose of 10 mrem/year to the maximally exposed off-site person. According to EPA calculations, done in the late 1980s, the maximally exposed individually from the MIT Reactor received a dose of 4 mrem/year and was located in an office building approximately 100 meters from the stack [54]. That calculation assumed a release of 4.2×10^3 Ci/yr. of Ar^{41} .

H.4 Analysis of Current Situation

There is little danger of a 10 MW MITR exceeding the EPA's standard of exposure of 10 mrem/yr. Even a 5 mrem/yr. standard would be easily met. Therefore, the current 5 MW MITR-II also easily meets the EPA standards.

The 5 MW MITR-II, assuming that 1993 is an accurate representation of future effluent production, should also easily meet the new 10CFR20 standards. However, meeting the NRC standards for a 10 MW MITR could prove to be more difficult if more reductions are anticipated in the limits within the next decade. As was shown in Table H-1, the projected effluents would be at about 73 % of the limits. This could be considered too close for comfort. Therefore, the conclusion of this analysis is that steps need to be taken to ensure that a 10 MW MITR-III would comply with NRC regulations with a bigger margin to allow for potential further reductions in the limits. Suggested improvements are included in the following section.

H.5 Potential Improvements

This section will examine several methods by which a higher level of confidence in the ability to meet the standards can be made

Dilution Factor

Because the gaseous effluents of the MITR are released from a stack which is 150 feet high and the concentration that leaves the stack is far higher than the concentration that any one individual would receive in any one place, the facility is allowed a dilution factor.

In 1959 the facility was granted a dilution factor of 3000. Hence, the facility is allowed to emit a radionuclide concentration of 3000 times the standards that any individual may receive. The initial calculations were based on Sutton's formula [55]. The formula, which *Meteorology and Atomic Energy* described as "the most widely used formula" of the various ones in existence at the time is a rather simple one. Using this formula a value of about 3200 for the dilution factor was found. To be conservative, this was reduced to 3000.

When the license was renewed in the 1970s, these calculations were again used. However, they were buttressed with other more sophisticated calculations. These used variants of the Gaussian Diffusion Model as well as real site conditions. While the Sutton's formula calculation was very general, the new calculations took into account

actual building heights. These calculations resulted in values in the range of an order of magnitude higher than the 3000 value. However, their intent was only to prove that the 3000 figure was conservative, not an attempt to raise the dilution factor. Given that these calculations show the 3000 dilution factor to be a very conservative estimate and that the remote monitoring stations report values that are lower than even the stringent EPA standards, one solution to the current situation would be to attempt to raise the dilution factor. In this study, several methods of calculating atmospheric transport of radionuclides were reviewed. Among these methods, the Particle-in-Cell (PICK) method was selected as the preferable option and should be investigated further.

Reduction of Source Terms

The best way to meet the standards would be to reduce the source terms. This would be most in keeping with the ALARA principle and would not require any technical specification changes or systems to remove the radionuclides from the flow to the stack.

As mentioned earlier, the MITR-II has made strides in the past decade in regard to reducing the amount of Argon-41 it emits. As the reactor is undergoing redesign, considerations should be taken to reduce the airflow through the reactor and seals should be made tighter to prevent air flow. In addition, every attempt should be made to ensure that fuel contains as few impurities on its surface as possible.

While reducing the source term is the preferred method, there are limits to what can be done. With all the work that has already been done, it is not certain that further large reductions in Argon-41 emissions can be achieved. Reducing source terms should always be a goal, but in all likelihood, it will not be enough to solve the problem.

Reduction of Releases

Another option to reduce the release would be to install an emission reduction system. A conceptual design for an off-gas treatment system was designed in-house [56]. In addition, there are various commercial treatment methods. Off-gas systems are fundamentally based on the delay of such releases so that a significant fraction of the activity may decay prior to the release.

The recommended system for the MITR-II was an adsorption system using charcoal. According to the in-house design estimates, releases of Kr and Xe isotopes could be reduced by a factor ranging from 4 to 325 depending upon the design. There would also be a reduction in the Argon-41 release, but it would be minimal.

Another potential treatment system would be a hold-up tank. This would be particularly good in that Argon-41-- the largest effluent-- has a relatively short half-life of 1.83 hours. The drawback, in the case of the MITR, is the large flow of air (8440 CFM) which would require a large and expensive tank.

Another way to deal with Ar^{41} would be to design hold-up systems in areas close to the source term. These would have much smaller flow rates, and thus would be more economical. Such a system would likely consist of a dual tank/compressor system. This option should be studied further.

H.6 Other Considerations for a 10 MW MITR

A consequence of upgrading to 10 MW would be a change in the reporting of effluent releases. The upgrade would place the MITR in a higher release category, thus increasing the reporting requirements. Currently, the annual releases of H^3 and Ar^{41} are reported to the NRC. These and the remainder of the effluents are reported to the MITR-II Safeguards Committee. At 10 MW, the release of all effluents would be reported to the NRC. Since these releases are projected to meet the standards, there should not be a problem. However, such reporting would make these releases a matter of public record, which would open them to public scrutiny.

Appendix I: Thermal-Hydraulic Study of the Mixing Pool [57]

The goal of this study is to investigate forced and natural convection of the coolant in the MIT reactor core tank during transient operation and to extrapolate the results to operating power levels up to 10 MW. Progress so far includes measurement of temperature distribution in the core tank for both forced convection and natural convection conditions and computational fluid dynamics simulation using the FLUENT code. A model will be constructed for the MITR-II and benchmarked by experimental results. This model can be applied to MITR-III transients because no major changes will be made to the core tank.

Operating conditions of interest are:

(a) Forced convection at constant flow rate

Coolant temperature in the core tank will change with time after a power change due to mixing of the flow from the core outlet with the coolant in the remaining of the core tank. The time delay of the temperature change at the hot leg in response to the power change will be measured and analyzed. The result will be used to determine the scram setting of the hot leg temperature sensor.

(b) Forced convection of the flow during pump coast down

Mixing becomes less efficient due to the decreasing flow rate. The flow rate decrease can be obtained from a momentum equation describing pump inertia and loop inertia. A neutronic model has to be included because the rod drop time (~ 0.5 sec. for 80% rod insertion [58]) may not be negligible and the increase of the coolant temperature might affect the thermal power output of the core due to temperature reactivity feedback.

(c) Natural circulation of the flow with natural convection valves open

When the flow decreases to a certain point, the natural convection valves will open due to the decreasing pressure difference between the core tank (top of the valve) and the cold leg (bottom of the valve). A natural circulation loop is then formed by core end—inside of flow guide(riser)—core tank (heat sink)—outside of flow guide (downcomer). The gravity force will drive the flow through the core, so the heat produced from the core can be removed passively. The mixing of the coolant will then be re-established.

I.1 Background Information

Several studies have been made to evaluate the natural convective cooling during loss of flow and reactor shutdown for the MITR [6,17,21,59]. However, most of them were done for the earlier core tank—before the installation of the flow guide, and an

instantaneous coolant mixing assumption was made to simplify the problem. In reality, the rise of the hot coolant plume from the flow guide during a loss of flow transient determines the driving force of the natural circulation and thus affects the flow rate.

A turbulent flow model needs to be used to model the plume rise and the coolant mixing in the core tank. A two-dimensional turbulent flow model will be developed for this study because the asymmetrical arrangement of the natural convection valves seems to have little effect on the temperature distribution (see section I.2). The coolant mixing model will be validated by comparing the results with those generated from the FLUENT code. This model will later be combined with the TAMLOC code which models the coolant system, except the core tank region, in detail and the combined model will be benchmarked using experimental data.

I.2 Thermal-Hydraulic Mixing Experiment

The purpose of the thermal-hydraulic mixing experiment is to investigate the coolant mixing in the MITR core tank during reactor scram at full flow and during a pump coast down transient. In the first part of the experiment, the reactor was tripped while the flow rate kept constant. In the second part of experiment, pumps were stopped right after the reactor is tripped. Natural convection valves were then opened, and natural convection flow established to remove the decay heat through the core. Temperatures at different locations in the core tank were measured versus time for both parts of the experiment. The results of the experiment will be used to benchmark an MITR core tank turbulent model capable of simulating turbulent mixing in the core tank.

I.2.1 Experiment Setup

Twenty thermocouples were located at different positions in the core tank to measure the temperature changes during the experiment. The thermocouples were affixed to two aluminum bars using aluminum channels. The aluminum bars were supported to the BCCL bridge, which was built for the loop experiment, at the top of the reactor core tank.

1/16" O.D., 304 stainless steel sheathed, ungrounded Chromel-Alumel (K-type) thermocouples were chosen for this experiment. The stainless steel sheath provided strength in order to avoid breakage of the thermocouples in the flow. The junction of this type of thermocouple is isolated from the protective sheath, so as to prevent interference from possible electric noise present in the core tank.

Ten thermocouples are distributed evenly on each of the aluminum bar. The axial locations of the thermocouples are shown in Figure I-1. The radial locations of the aluminum bars in the core tank are shown in Figure I-2. At each station where the temperature is to be measured, the tip of a thermocouple sticks through the aluminum bar and makes contact with the bulk coolant. The layout of the first thermocouple station (TC-1) is shown in Figure I-3. The signals from the thermocouples were read by an Hewlett-Packard (HP) 3852A data acquisition/control unit. The data acquisition unit can

Figure I-1 Thermocouple Locations

Figure I-2. Radial Positions of the Aluminum Bars Used to Fix the Thermocouples

Figure I-3. Cross-Section View of Thermocouple Location TC-1 (not to scale)

be programmed for K-type thermocouples to convert voltage to temperature directly. The temperature readings were then sent to a hard disk drive of an HP computer for storage. The frequency of reading the temperatures is about one second each cycle, which is limited by the speed of the computer. A computer program was written using HP BASIC language, which served as an interface between the data acquisition unit and the storage unit and controlled the initiation, termination and the frequency of the data recording.

The thermocouples were calibrated at 0°C (in iced water), room temperature, and 100°C (in boiling water) using a thermometer before assembled to the aluminum bars. The temperature variation is within 0.5°C. The thermocouples were calibrated again using the HP data acquisition unit after assembly and welding. The temperature variation versus time becomes large (up to 1°C at 0°C, 2°C at 100°C) which is possibly due to the unfiltered electronic noise of the system.

Two other temperature measurements were made at the outlet of the core tank. The temperature sensors were previously installed by the reactor staff. These signals of these temperature sensors are amplified voltages and were recorded by an IBM compatible computer. The core inlet temperature has to be recorded manually during the experiment because the signal of the temperature sensor cannot be read by the data acquisition unit.

I.2.2 Experimental Results

Fourteen sets of data were collected during two weeks with the assistance of the reactor operation crew. A list of the experiment is shown in Table I-1. Primary coolant was kept at full flow for P, F, and R runs while the reactor power changes in different ways. These runs are to be used to benchmark forced convection modeling from which the response time of the temperature sensor at the outlet of the core tank can be derived. Secondary coolant, shield coolant, and D₂O temperatures were also recorded manually from the control room for P-runs. These experimental data can be later used to benchmark computer codes simulating the MITR coolant system, e.g., TAMLOC and MITR Simulator.

The experiment results show favorable temperature distribution in the core tank during transient operations. For natural convection cases (Z-runs), experimental results imply that the hot coolant plume rises to the top of the core tank because thermocouples TC-A1 and TC-B1 measured the highest temperature when the coolant in the core tank started to heat up. For run Z-3, the temperature difference between the top and bottom of the core tank is about constant (~ 3 °C) after 4 minutes. The time constant of the mixing for the bottom of the core tank to reach the same temperature of the top is about 400 seconds. The temperature decrease at the first 2 minutes of the transient is due to the combined effect of pump inertia and power decrease from 4.25 MW to decay power level. The rate of power decrease is larger than that of flow (to the heat exchangers) decrease and that results in a net effect of core tank temperature decrease. Then the

Table I-1
List of Runs of the Thermal Hydraulic Experiment

Run ID	Description	Power Level	Duration of Steady-State Operation	Data Recorded
P-1	full flow	1 to 2	~ 20 min. @ 1 MW	• Temperatures from recorder
P-2	full flow	2 to 3	~ 20 min. @ 2 MW	• MT-1
P-3	full flow	3 to 4	~ 20 min. @ 3 MW	• Core tank temp*
P-4	full flow	4 to 4.5	~ 20 min. @ 4 MW	• MT-1
F-31	scram in full flow	4.5	~ 10 hours	• Primary inlet temps.
F-32	scram in full flow	4.5	~ 12 hours	• Core tank temps*
F-1	scram in full flow	2.5	~ 12 hours	• MT-1
Z-3	scram and trip pump	4.25	~ 5 hrs @ 4.5 MW ~ 7 hrs @ 4.25 MW	• Core tank temps*
Z-2	scram and trip pump	3.25	~ 7 hours	• MT-1
Z-4	scram and trip pump	4.9	~ 12 hours	• Core tank temps*
R-4	full flow	4.9 to 4	~ 20 min. @ 4.9 MW	• T _{inlet} from recorder
R-3	full flow	4 to 3	~ 20 min. @ 4 MW	• MT-1
R-2	full flow	3 to 2	~ 20 min. @ 3 MW	• Core tank temps*
R-1	full flow	2 to 1	~ 20 min. @ 2 MW	

temperatures start to increase because the coolant in the core tank becomes the only heat sink to remove the decay heat.

Experimental results for run F-3 shows that temperatures measured by TC-A1 and TC-B1 (located at the top of the aluminum bars) are very close to each other and are all higher than the others. Also they are about the same as the temperature measured at the outlet of the core tank. The result suggests that the upward momentum of the coolant is capable of driving the plume to the top of the core tank, despite the suction force of the primary flow. However, this needs further investigation because the temperature sensor at the core tank outlet was calibrated separately by the reactor staff. The mixing time constant observed from the result for this case is about 50 seconds.

All the results shown in this section are curved fitted from scattered data points. Further data analysis is needed to filter the electronic noise and determine the range of uncertainty caused by measurement only.

I.3 Theoretical Confirmation Using FLUENT Code

FLUENT is a general purpose computer program for modeling fluid flow. It was made accessible to the users by means of an interactive menu-driven interface for problem definition, computation and built-in graphics. A pre-processor named PREBFC can be used to setup geometry and grid before using FLUENT for calculation. The PREBFC package works better for complex geometry setup and works similar to a CAD. The current study uses PREBFC and FLUENT v4.1 on CRAY X-MP running UNICOS 6.1.4 at MIT. Due to the limitations of the FLUENT code, the core tank can only be analyzed by separating it from the reactor core and the primary loop and the secondary loop, i.e., the flow in the core tank has to be decoupled from the primary loop and the secondary loop.

The loss coefficients of the natural circulation valves were also analyzed for both downflow and upflow cases by running this code. It is hoped that by predicting the flow and the pressure changes through the valve directly through the computer code, a more reliable result can be obtained for future uses.

I.3.1 Thermal-Hydraulic Mixing in the Core tank

Control mechanisms, fuel storage rack and other instruments in the core tank are neglected in the prediction of the forced convection and natural convection flow in the core tank. The coolant-atmosphere interface at the top of the core tank is simulated as a wall boundary condition. The validity of this assumption will be discussed later.

Forced Convection

Steady-state analysis of forced convection is done for both 2-D and 3-D core tank. For the 2-D case, it is assumed that the core tank is axisymmetrical and the outlet pipes are thus reduced to a ring around the core tank. The outlet area is conserved so that the

momentum exiting the core tank remains constant. For the 3-D case, only half of the core tank is modeled. Three outlet pipes are modified as a slot, which extends 90 degrees on the core tank. The outlet area and exit momentum are also conserved in this case.

Figure I-4, I-5 and I-6 are the graphical output of FLUENT for the 2-D case. The boundary condition at the outlet of the flow guide (labeled INLET on the figures) corresponds to a flow rate of 2050 gpm (rated flow rate of the MITR-II). Figure I-5 shows that the flow at the top of the core tank traveled directly from the outlet of the flow guide, which then goes down to the outlet. A swirl is found in the middle left part of the core tank as shown in Fig. I-6. It is formed due to the driving forces of the flow near the wall (downward) and the inlet flow (upward), which results in the local circulation of the swirl.

Figure I-7 shows the velocity distribution predicted for the 3-D core tank. There is only one plane shown in the figure due to the complicated geometry. The positions of the temperatures measured in the experiment are approximately on this plane. The result shows that the predicted upward velocity in the 3-D case is larger than that in the 2-D cases. This is because the suction force of the outlet is relatively smaller in the 3-D case, which allows the flow to travel further up to the top of the core tank. The effect of reducing 3-D geometry to 2-D needs to be studied further.

Transient analysis is done for the forced convection case to see how the temperature changes in the core tank when a step change of the reactor power occurs. For simplicity, it is assumed that a 2°C temperature step change occur at the inlet of the core tank; this temperature change corresponds to a power change of about 1 MW for the MITR-II.

Figure I-8 is the predicted temperature at the center of the outlet (from the core tank). For rapid convergence, the velocities and pressures are frozen during the transient calculation. That is, the velocities and pressure distributions are assumed the same as those predicted in the steady-state analysis. For the first 18 seconds, the FLUENT code calculated enthalpy, temperature and turbulent kinetic energy to obtain the converged results. Due to the limited CPU time available and also the slow convergence, the turbulent kinetic energy is then frozen beyond that time. Figure I-9 is the result of the same transient as above, but with a higher flow rate (corresponding to 2800 gpm, predicted flow rate for the 10 MW MITR).

It seems that the response time does not improve significantly when the flow rate increases from 2050 gpm to 2800 gpm. However, the temperature distribution in the core tank shows a better mixing at the lower left region of the core tank for the high flow rate case.

Natural Convection

Steady-state analysis of the natural convection is done for 2-D geometry, the flow rate is assume 1% of the rated flow rate of the MITR-II. The four natural convection

Figure I-4 Velocity Distribution of MITR Forced Convection

Figure I-5. Velocity Distribution of MITR Forced Convection (magnified view-1)

Figure I-6. Velocity Distribution of MITR Forced Convection (magnified view-2)

Figure I-7. Velocity Distribution of MITR Forced Convection in 3-D Core Tank

Figure I-8. Temperature at the Center of the Hot Leg Predicted for the 2050 gpm Case.

Figure I-9. Temperature at the Center of the Hot Leg Predicted for the 2800 gpm Case.

valves are reduced to a "ring" of flow outlet, with the outlet area and the momentum are conserved. Figure I-10, I-11, and I-12 are the result of FLUENT prediction.

An attempt of a transient analysis failed due to the following reasons:

- 1) The driving force for the natural convection (coolant heating in the core) cannot be modeled due to the limitation of the computer code.
- 2) An artificial boundary condition, which sets the pressure difference at the inlet and the outlet of the natural convection flow, have caused rapid divergence. The reason of the divergence is still unresolved.

I.3.2 Loss Coefficients of the Natural Convection Valves

Figure I-13 shows the current design of the natural convection valve. The loss coefficient of the natural convection valve was estimated previously by dividing the valve into several sections (in each section the flow area assumed constant), then the loss coefficients are added up due to contraction and expansion with respect to the same reference location. However, the distributions of the velocity at the inlet and outlet of the flow have significant effect on the prediction of the loss coefficient. That is, if the flow is not fully developed, the pressure loss cause by contraction and expansion will be larger than expected [33,60].

The FLUENT is used to calculate the loss coefficients of the natural convection valves for both upflow and downflow. For downflow cases, five inlet velocities (to the natural convection valve from the top) were evaluated which correspond to a range of flows between 1% to 10% of the rated flow rate assuming all four valves are open (natural circulation flow is about 1% of the rated flow rate according to Reference [21]). This range of flow rates should cover the natural convection flow rates for 5 MW and 10 MW designs.

At the initial stage of the establishment of natural convection flow, upflow will occur when the aluminum ball starts to drop. The flow rate is then determined by a momentum equation. Three upflow cases were evaluated, the inlet velocities of the upflow corresponding to 1%, 5% and 10% of the rated flow rates assuming all four valves are open. All the flow rates analyzed are in the turbulent flow regime ($Re > 2000$), k-e model (with turbulent intensity equals ten and the characteristic length equals the radius of the flow guide) is used to calculate the turbulent quantities.

The output of the computer code includes relative static pressure, U velocity (z-direction) and V velocity (r-direction) for each cell location. The static pressure difference at the inlet and outlet of the valve is thus calculated as follows.

$$\Delta P = P_{in} - P_{out} = \frac{1}{A_{in}} \sum_i P_i A_i - \frac{1}{A_{out}} \sum_j P_j A_j \quad (\text{Eqs. I-1})$$

where i and j correspond to the cells at the inlet and the outlet area.

Figure I-10. Velocity Distribution of MITR Natural Convection

Figure I-11. Velocity Distribution of MITR Natural Convection (magnified view-1)

Figure I-12. Velocity Distribution of MITR Natural Convection (magnified view-2)

Figure I-13. Current Design of the Natural Convection Valve [21]

The pressure loss formulated above includes form loss, kinetic energy loss/gain and friction loss within the valve. According to the definition of form loss, the kinetic energy change should not be included in the pressure loss [60].

Comparison of the calculated loss coefficient with previously estimated values is shown in Table I-2. Now the loss coefficient is based on the flow velocity at the 4" cross section in the valve and is independent of the number of valves open to the natural convection flow. The new prediction (using the FLUENT code) includes the friction loss in the valve (which is less important than the form loss). The previous predictions only estimated the form loss and assumed that the friction in the valve is negligible compared to the form loss.

Table I-2
Comparison of Calculated Loss Coefficients

Author	Upflow	Downflow
Afonso-Morales (1972)*	N.A.	$K_{ref} = 5.55$
Shaffer (1976)*	N.A.	$K_{ref} = 7.08$
Bamdad-Haghighi (1977)	$K_{ref} = 4.59$	$K_{ref} = 12.71$ (turbulent) $K_{ref} = 15.09$ (laminar)
Current Study (1994)	$K_{ref} = 46.64$	$K_{ref} = 51.95$

* analysis done for the earlier design.

The new prediction of the loss coefficients are larger than the previous ones by a factor of four for the case of downflow and by a factor of ten for the case of upflow. This discrepancy is mainly due to the following reasons:

- 1) Pressure drop is larger if the flow is not well-developed. This effect was not taken into account in the previous predictions.
- 2) The velocity distributions upstream and downstream of the contraction/expansion play an important role in the prediction of the loss coefficients. The velocity distribution parameter is defined as

$$K_d = \frac{1}{AV_{avg}^2} \int_0^A V^2 dA \quad (\text{Eqs. I-2})$$

The effect of the velocity distribution parameter (downstream) to the prediction of the contraction loss coefficient is shown in Fig. I-14, the velocity distribution upstream is assumed uniform in this case. The results from FLUENT have shown that the high velocity region is located between the top of the valve and the bottom of the aluminum ball, where the flow area changes continuously. So that this region contributes most of the pressure drop through the valve.

It would appear that the 5 MW transient results are not too sensitive to this evaluation of K_{ref} since the experimental results for natural circulation cooling have been satisfactorily demonstrated to be in the expected temperature range.

Figure I-14 Effect of velocity distribution parameter to the predicted coefficient

I.4 Future Work

A two-dimensional k-e turbulent model is to be developed for the MITR core tank for both natural convection and forced convection cases. Although there are computational fluid dynamics (CFD) codes available commercially like the FLUENT code, there are some drawbacks in using them. For example, the FLUENT code cannot handle multi-channel cases, e.g., the reactor coolant channels. And since the inlet and outlet boundary conditions of the core tank is coupled to the primary and secondary loops, it is necessary to combine the core tank model to the coolant loop model in order to simulate the response of the coolant system. It will be worthwhile to construct a new model which is to be combined with the coolant model rather than supplying a long "user defined subroutine" in the CFD codes to define the primary and secondary loops for the core tank model.

Once the turbulent model is benchmarked by the results of the FLUENT runs described in the last section, the coolant loop model will be constructed based on the TAMLOC code [6]. Now the coolant system model can be benchmarked by the experimental data obtained for various transients.

The overall coolant system model (includes core tank, primary loop and secondary loop) will be used for the following purposes.

- 1) Predict transients for the 5 MW MITR.
- 2) Predict transients for the 10 MW MITR.
- 3) Examine the effect of flow guide height and natural convection valve designs to natural convection flows.

I.5 Application to Industry

Commercial reactor designs of the next generation emphasize safety by means of improved safety system reliability and performance. Applying safety systems using natural forces has then become one of the major subjects of studies. For example, SBWR design uses natural convection in the reactor vessel to remove heat from the core during normal operation and natural circulation to remove decay heat from the reactor core during transients. The same concepts have also been used in other types of advanced reactors like MHTGR and PRISM (liquid metal cooled fast reactor).

The passive safety concept has been applied in the MIT reactor for nearly twenty years, design and operation experience has since accumulated. With the advent of new computer technology, now it is possible to simulate fluid flow in detail, i.e., turbulent mixing, on a work station or personal computer. It is hoped that a compact, while powerful, model can be developed for the whole reactor cooling system of the MITR. This experience can later be used by the nuclear industry for designing a passive safe system.

J.1 Introduction

Nuclear reactor accident analyses have traditionally been performed with many parameters set equal to extreme or "hot channel" values to account for uncertainties. With this approach, it is likely for calculated results to give conservative representations of reactor safety. However, this approach has deficiencies:

- First, the accumulation of conservatism may give an unrealistic pessimistic final result. This results may cause designers or operators to make inappropriate decisions.
- Second, by entering conservatism at many locations in a calculation, the conservatism of the results cannot be easily evaluated.

More recent approaches to nuclear reactor thermal hydraulic analyses have incorporated statistical evaluations of uncertainty for a variety of parameters (correlations, properties, dimensions, operating ranges, etc.). The approaches have included "best estimate plus uncertainty" calculations. The efforts and information required to execute the changes are far from trivial.

The present study aimed at quantifying the uncertainties in some key parameters, for use in ongoing design studies to upgrade and relicense the MITR.

The transient under investigation is a pump coastdown. For this transient, a Phenomena Identification and Ranking Table (PIRT) has been developed to reduce the uncertainty study to the most important phenomena. A group of three experts (Prof. Meyer, Prof. Henry, and Prof. Lanning) have developed a table of phenomena and have ranked it on a scale of from 1 (least important) to 5 (most important). From this table, three important phenomena were selected for a preliminary study:

- 1) The time to scram for the scram signal
- 2) The time to scram for the control blade insertion
- 3) The flow instability prediction

The study of the uncertainty of each phenomenon is summarized in the following sections.

J.2 Scram Signal

During a pump coastdown transient, the coolant flow suddenly decreases from the initial flow to a nearly zero flow. The temporal behavior of the coolant flow rate is of prime interest for the study of a pump coastdown transient uncertainty.

A set of four experiments has been performed to measure the flow as a function of time [22] (see Figure J-1). The noisy voltage data have been directly recorded and must be regressed on to a parametric function. The function depends on the initial flow rate; the initial flow rate during the flow measurements was constant at the nominal flow rate (2050 gpm). The parametric voltage function will then be converted into a flow function using a conversion proposed by Terry [22].

The main uncertainty in the data available comes from the measurement device that causes a noise to be added to the measurement of the voltage. There is also a small uncertainty in the linear conversion from voltage to flow and in the setting of the initial flow rate.

Figure J-1. Flow Coastdown Data

The parametric shape of the flow coastdown curve is not known. The following statistical model

$$\log(V_i(t_j)) = f(t) + \varepsilon_j \quad (\text{Eqs. J-1})$$

where the parametric function $f(t) = a + bt + c\exp(-t) + d\exp(-t^2)$ has been selected for its nearly unbiased nature and constant variance after 5 seconds; it is however biased for times between 0 and 5 seconds. A better parametric function should be proposed to achieve the unbiased property.

The best estimate function $f(t)$ can be estimated by least squared regression on the average of the four experiments. The uncertainty could then be estimated by doing individual regressions $f_i(t)$ for each experiment and by estimating the variance $\sigma^2(t)$ as a function of time. The distribution of $f(t)$ could then be assumed to be approximately Student with 3 degrees of freedom. However, an unbiased parametric function should be developed first.

The best estimate function based on the experiments is as follows:

$$f(t) = \exp(-1.87 + 0.41t + 2.95\exp(-t) - 0.68\exp(-t^2)) \quad (\text{Eqs. J-2})$$

Because the function is biased, a bias correction should be applied before estimating the variance. Hence, a better parametric model should be developed.

J.3 Control Blade Insertion Time

To control the core reactivity, the MITR-II has six control blades and a single regulating rod that is of no concern here. During normal operation, the amount inserted is about the same for all control blades (they must be kept no more than one inch apart) and is adjusted over long intervals to account for burnup and xenon. Each blade is maintained to a withdrawn position by a magnetic field that collapses after a trip signal is detected. For a flow coastdown transient, the signal to scram is sent by the fastest of three independent flow detectors. Each blade falls independently of one another in its individual channel; however, the reactivity is mainly controlled by the fastest control blade.

There are three important phases to consider:

- 1) Before the time $t = 0$ when the transient starts;
- 2) Until the time T_f when the magnetic field receives the signal; and
- 3) After T_f .

By looking at the time to scram records of the six blades, an "average" blade has been selected to represent the behavior of all the blades. The data available indicate the

time to reach 80 % insertion. Blade 3 has been selected as the "average" blade because its time to reach 80 % insertion was the lower median of the six blades. Therefore, the uncertainty of the following function was evaluated based on the measurements made on blade 3.

$$L(t) = \begin{cases} L_o & t \leq T_f \\ L_o + D(t - T_f) & t > T_f \end{cases} \quad (\text{Eqs. J-3})$$

where,

L_o represents the initial level of insertion during normal operation.

$D(t-T_f)$ represents the amount dropped as a function of time starting at time T_f .

T_f represents the time for the first of the three flow detectors to detect the flow coastdown and send the signal for the magnetic field to collapse.

J.3.1 Initial Level L_o

Data are available at every hour over a year. One can use these data to determine a density function that fits the empirical density best. However, the judgment of a reactor operator was used to estimate the density function and is shown in tabular form below.

Level of Insertion	Percentage of the Time
0-8"	0 %
8-12"	30 %
12-15"	35 %
15-17"	30 %
17-18"	5 %
18-21"	0 %

At a 95 % confidence level the initial level of insertion, L_o , is 8.67 inches.

J.3.2 Detector Response Time T_f

The three flow detectors MF1, MP-6, MP6A were recorded five times for a flow coastdown starting at the nominal flow (2050 gpm).

Detector	Time (s)					Average
MF-1	0.66	0.64	0.63	0.63	0.64	0.64
MP-6	0.28	0.28	0.27	0.28	0.26	0.27
MP-6A	0.33	0.34	0.33	0.33	0.32	0.33

These data were recorded within few minute intervals and the standard deviation accounts only for measurement errors. However, the detectors are readjusted every year so that the uncertainty is larger from year to year.

These data agree reasonably well with a Normal distribution. Hence, one can say that T_f is Student distributed with 2 degrees of freedom with mean of 0.41 seconds and standard deviation 0.12 seconds.

J.3.3 Amount Dropped D

The function of interest is the level of the blades as a function of time. With a α -level of confidence, this function is as follows:

$$D(t) = \left\{ \begin{array}{ll} 0 & v \leq 104 \pm z_{\alpha/2} 5.04 \\ \frac{1}{2(625 \pm z_{\alpha/2} 8.94)^2} (1000v - (104 \pm z_{\alpha/2} 5.04))^2 & \text{otherwise} \end{array} \right\} \quad (\text{Eqs. J-4})$$

which is plotted on the following figure for a 95 % confidence level.

Figure J-2. Blades Position as a Function of Time with a 95 % Confidence Level

J.4 Flow Instability Prediction

It has been shown by Whittle and Forgan [42] that the onset of flow instability (OFI) can be well characterized by the ratio Q of the channel power to the power to saturation.

$$Q = \frac{4q''_{avg} L}{Gc_p D(T_s - T_{in})} \quad (\text{Eqs. J-5})$$

The flow rate is stable if the ratio Q is less than a critical ratio Q_c calculated from experimental data at the OFI. That is, the flow rate is stable if

$$S_{OFI} = \frac{Q_c}{Q} > 1 \quad (\text{Eqs. J-6})$$

The ratio Q has been selected as the quantity used to detect OFI after Whittle and Forgan noticed the quasi-constant property of Q_c over a wide range of conditions such as power shape, pressure, or temperature. However, they have found that Q_c varies with the L/D ratio where L is the length of the channel and D is its hydraulic diameter.

$$Q_c = \frac{1}{1 + \eta \frac{D}{L}} \quad (\text{Eqs. J-7})$$

Based on a set of experiments, they estimated $\eta = 25$.

Saha and Zuber [27] worked on the onset of significant void (OSV). This phenomenon appears before OFI. Their correlation is based on a relation between Peclet number

$$Pe(z) = \frac{GDc_p(z)}{k_f(z)} \quad (\text{Eqs. J-8})$$

and Stanton number

$$St(z) = \frac{q''(z)}{Gc_p(z)(T_s(z) - T_b(z))} \quad (\text{Eqs. J-9})$$

Stanton number plotted versus Peclet number on a log-log scale at OSV over a large range of conditions reveals two regions of the X-Y plane; the lower region characterizes the state before OSV while the upper region that of post OSV conditions.

The important aspect of this study was to identify and quantify the uncertainty of the Stanton-Peclet OFI correlation so that the unstable region can be predicted with a certain

level of conservatism. This has been done based on the data found in References [42], [61], and [62] using the approach of Saha and Zuber.

By the least square method, the following correlations were developed:

$$St = \frac{0.0137}{Pe^{0.024}} \quad (\text{rectangular channel}) \quad (\text{Eqs. J-10})$$

$$St = \frac{0.1076}{Pe^{0.219}} \quad (\text{cylindrical channel}) \quad (\text{Eqs. J-11})$$

These correlations compare very well with the OSV correlation $St = 0.0446/Pe^{0.18}$ in Reference [35] and the one of Saha and Zuber.

POTENTIAL MITR IMPROVEMENTS

Potential upgrades in the design of the MITR were considered in the studies summarized in the following appendices.

Appendix K: Flux Synthesis Project [63]

K.1 Overview

The goals of the Instrumented Flux Synthesis Project are:

- 1) To experimentally confirm the Instrumented Flux Synthesis theory using data from the MIT Reactor, and
- 2) To design an automated controller based upon this theory.

Progress thus far has included:

- 1) Presentation of the theory as a Ph.D. thesis in 1991, in which the validity of the theory was confirmed using numerical modeling.
- 2) Gathering of the needed experimental data from the MITR core. Neutron flux measurements were taken under static and transient conditions.
- 3) Obtaining the proper neutronics codes for testing of the theory.

K.1.1 Motivation for Instrumented Synthesis Method [64]

Rates of heat and energy removal, though fundamental to the safe operation of a nuclear power plant, are not directly measurable. We rely upon such measurable quantities as neutron flux, temperature increase in the coolant, coolant pressure, and flow rate to tell us whether we are operating the plant safely while, at the same time, obtaining the maximum power possible from the facility. For optimal control of the reactor, we would like to ideally be able to do the following:

- 1) Compute the instantaneous, three dimensional distributions of neutron flux, power density, xenon concentration, coolant temperature, and coolant density, and infer from them actual values of reactivity, axial and radial peaking factors, axial offset, as well as thermal margins;
- 2) Compare the theoretical predictions of observable quantities with local measurements provided by fast-responding instrumentation; and
- 3) Act upon some control variable to achieve a desired, optimal plant response while maintaining all technical and safety constraints.

These points form the heart of an automated control system, in which the operators would carry out merely a supervisory role.

The crux of completely determining the state of the system is knowledge of the instantaneous neutron flux density throughout the reactor. Local power density is given

directly by neutron flux density, and the remaining quantities of interest can be inferred. Thus the center of an automated controller revolves around knowing the neutron flux density at all locations in the reactor.

Current methods of core design do give estimates of three-dimensional flux density under static and transient conditions. Cross sections and geometrical data are given as input to a diffusion theory or transport theory code, which is capable of finding the flux at all points in the reactor *based upon these data*. Transient situations are modeled by determining what thermal-hydraulic conditions exist, and using this information to effect changes in the neutronic calculations, which in turn changes the thermal-hydraulics. Such a feedback approach represents the standard method for analyzing transients. There are two major drawbacks to these approaches:

- 1) Both the static and transient methods rely upon the assumption that the situation being modeled is known exactly. In reality, changes in operating conditions result in complicated feedback effects, making modeling such transients difficult.
- 2) Though nodal codes are becoming increasing fast to run, they cannot model sophisticated transients in real time. Even if such speed were attained, they would still be limited by incomplete knowledge of the reactor conditions during the transient.

The instrumented method serves to eliminate these restrictions. It is capable of providing accurate values of the flux density in real-time without relying upon wholly accurate input data.

K.2 Theory

K.2.1 Derivation

The backbone of any synthesis theory is the premise that an unknown flux can be created (synthesized) by combining known, pre-computed expansion functions

$$\underline{\phi}(t) \approx \sum_{k=1}^K \underline{\Psi}^{(k)} T^{(k)}(t) = \underline{\Psi} \underline{T}(t) \quad (\text{Eqs. K-1})$$

In the above equation, the flux is synthesized from K expansion functions $\underline{\Psi}^{(k)}$, where $\underline{\Psi}^{(k)}$ is a column vector of length GN , $G = \#$ of groups, $N = \#$ of nodes. How much weight is given to a particular expansion function is determined by the mixing coefficients $T^{(k)}(t)$. Thus the $T^{(k)}(t)$ are the unknowns.

In traditional synthesis methods, those coefficients are found using a weighted residual method. In the instrumented synthesis method, the $T^{(k)}(t)$ are found using in-core instrumentation. Thus it reconstructs the flux from actual reactor data, and presents

an opportunity for accurate flux reconstruction in real time. From this flux reconstruction, kinetics parameters such as reactivity are inferred. The governing equation is given below.

$$\underline{A}\underline{T} \approx \underline{C}(t) \quad (\text{Eqs. K-2})$$

where $\underline{C}(t)$ is a vector containing the detector readings, $\underline{A}$ is a matrix containing inner products of response functions of the detector and the expansion functions, and $\underline{T}$ is a column vector of mixing coefficients. If $\underline{A}$ can be inverted, the $\underline{T}$, thus the $\underline{\phi}(t)$, are found.

K.2.2 Theoretical Proof

The instrumented synthesis theory has already been tested using computer modeling of reactor transients. The agreement between synthesis theory and nodal theory predictions was quite close. Below is a synoptic diagram of the numerical testing of the theory [64].

Figure K-1. Theoretical Testing of the Instrumented Synthesis Method

The neutronics code QUANDRY was used to provide the "real" answers against which the synthesis method was checked. It also provided, as input to the synthesis code, simulated detector counts, expansion functions, and a weight function for determination of

the kinetics parameters. To test the effects of noise, random fluctuations were added to the simulated detector readings, and then filtered by the synthesis code.

The power, flux, and kinetics parameters as determined by the synthesis theory was compared to those obtained in QUANDRY, and found to be in good agreement.

K.3 Experimental Confirmation

K.3.1 Current Outline for the Project

For a true test of the theory, application to an actual reactor system is needed. The MIT reactor has been used to gather data to challenge the theory's ability to reconstruct neutron fluxes in static and transient states. Below is a diagram of how those tests will be conducted. This can be compared to the numerical testing diagram in Figure K-1.

Figure K-2. Experimental Testing of the Instrumented Synthesis Method

Note the differences in the experimental and numerical testing:

- Most important, the detector readings are not simulated, but taken from actual measurements in the MITR core.
- The neutronics code used to obtain the flux shapes and weighting function is now VALE. This was selected for its ability to model triangular systems. This geometry is the most suitable for the MITR core.
- Noise already exists in the detectors. Thus noise will not be added, but will be filtered by the synthesis code.
- The perturbation is in the core, not the transient code. Thus the modeling by the transient code QUARTZ will be only approximate.

There are points to keep in mind when testing the instrumented synthesis method which make it an attractive option for reconstructing flux shapes:

- Cross sectional data in the synthesis code is needed only for the fission chambers. Even then, detailed data are not required. The only information needed is the current generated by a given fission chamber for a given flux.
- Cross sectional data is required to generate expansion functions (the $\Psi^{(k)}$). But the in-core flux is synthesized from several different flux shapes, none of which in general will be an exact match. Therefore, only approximate shapes, therefore approximate cross sections, are needed.
- The synthesis method is able to accommodate noise in the detection system, as well as inaccurate predictions in control rod worths, reactivity insertions, etc.

The most notable restriction in experimental validation of the method is this: the matrix $\underline{A}$ is, in general, not square and, therefore, not directly invertible. The method used to obtain an inversion of the matrix, the singular value decomposition method, works best if the system of equations is underdetermined, or more unknowns exist than equations. In other words, there should be more detectors than expansion functions. Nine detectors were used in the MITR tests. This is thought to be sufficient. Though this is a restriction in experimental confirmation, power reactors are generally rich in in-core instrumentation. No problems are foreseen in commercial application.

K.3.2 Experimental Data

The diagram below illustrates how the data were gathered from the MIT reactor core. Three fission chambers were arranged axially in three water vent holes, for a total of nine fission chamber detectors. The fission chambers are lined with U^{235} . This uranium will fission when subjected to a flux of thermal neutrons, and the charge carried by the fission products is collected in the chamber, creating a current. This current gives a

measure of the thermal neutron flux at the location of the fission chamber. The chambers are selected above other types of detectors because of their fast response times to changes in neutron flux. A quick response is needed to adequately track fast transients.

Figure K-3. Schematic of Digital Data Acquisition System

The signals from the detectors were fed through switchboxes to ammeters, which measured the currents. The measurements were recorded by a computer for later review. Two of the recorded transients are shown in Figures K-4 and K-5. The current is directly proportional to the neutron flux at that point.

Withdrawal of Shim Blade #6 Detectors at Core Midplane

Figure K-4. Current from Fission Chamber Detectors During Power Increase

Drop of Shim Blade #1 Followed by Reactor Scram

Figure K-5. Current from Fission Chamber Detectors during Shim Blade Drop

The measurements have been checked for consistency. The components of the setup remain in the reactor containment if more data are needed.

K.3.3 Confirmation

Work now shifts to the numerical work outlined in Figure K-2. Codes have been acquired to model the MITR core, providing flux shapes for use in the synthesis code. This modeling is being done in conjunction with the fuel management portion of the MITR-III group. This combination of resources should speed the process in both areas.

K.4 Application to MITR-III and Industry

The redesign of the MIT reactor can certainly proceed independently of the Instrumented Synthesis Project. But the potential of the Instrumented Synthesis Theory to serve as the backbone of an automated controller may help to make the MITR-III a valuable testing ground. It was suggested at the beginning of the 1993-94 academic year that the Synthesis Project be considered when designing the MITR-III. The suggestion was made that, if the synthesis method is to be properly applied, in-core instrumentation is required. Such instrumentation would have to be placed into the MITR-III design. It is hoped that eventually, the MITR-III could be used to test the complete automation of reactor control, in which the Instrumented Synthesis Method would play a central role.

Because the MITR is quite different in design and mode of operation than commercial power reactors, the question may be legitimately asked "Can the method be transferred to commercial plants?" The answer is yes, and in some ways the application is easier than in the MITR. There typically is much more instrumentation in commercial reactors. For example, BWR's already have extensive in-core instrumentation. All that is needed to incorporate the method is to have the signals fed to an additional computer. Such ease of operation makes the method an attractive option for utilities. There is no extensive retro-fitting needed, and the expense would be practically negligible.

As commercial power plants move to greater automation in control, the instrumented synthesis method is likely to play a significant role.

Appendix L: Fission Converter-Based Epithermal Neutron Beam for Future BNCT Facility [65]

The collaborative research program in BNCT between Massachusetts Institute of Technology and New England Medical Center is currently ready to begin Phase I clinical trials of BNCT for the treatment of subcutaneous peripheral melanoma using the current MITR-II medical beam [66]. If these and other planned clinical trials prove to be successful, a stronger beam capable of rapidly treating many patients per day would be necessary. Rief has shown that a fission converter driven by a nuclear reactor can provide a very clean and highly intense beam of epithermal neutrons [67].

The spring 1994 MIT course 22.33/033 Nuclear Engineering Design class examined the design of a facility to meet this need, a fission converter-based epithermal neutron beam located in the current MITR-II thermal column [65]. A view of the proposed facility is shown in Figure L-1. Because the neutronics design of the beam filter and collimator is the subject of ongoing thesis research, this project focused on neutronics and thermal hydraulics design of the fission converter.

L.1 Fission Converter Design

Our design for a high intensity high quality epithermal beam employs a fission converter source of fast neutrons which is filtered and moderated to more favorable energies by resonance scattering materials and then collimated onto the patient position in a shielded medical irradiation room. In the design, the fission converter, which consists of a row of 12 MITR-II (spent) fuel elements, is located in a vertical duct of the thermal column and is centered on the 14 in. window through the graphite reflector which "looks" toward the core. The fission converter is enclosed in a redundant aluminum tank which "hugs" the fuel to minimize the amount of coolant water around the fuel.

The redundant tank and anti-siphon valves ensure that the fuel is kept covered in a credible LOCA. Passive shutdown cooling to remove heat is accomplished with natural circulation valves. The design should be relatively easy to license since it involves relatively low power densities. A simple fuel handling plan has been developed which requires a connector piece that allows the fuel transfer flask to be fitted to the thermal column. The instrumentation and controls required for such a facility have also been examined.

L.2 Neutronic Analysis

A Monte Carlo model of the coupled fission converter beam [68] and MITR-II core [25] implemented in the code MCNP [10] was used for neutronic analysis. The Monte Carlo model provided fission converter power distributions and neutron flux at the patient position. Using burned fuel in the fission converter and light water cooling with an aluminum/aluminum fluoride filter, the fission converter provides an epithermal flux of

Figure L-1. Conceptual Design of Fission Converter-Based Medical Irradiation Facility

about 2×10^{10} n/cm²s with a fast to epithermal flux ratio of 0.118 ± 0.002 at the patient position and produces 108.3 ± 0.2 kW fission power. The same configuration with heavy water cooling provides an epithermal flux of about 3×10^{10} n/cm²s with a fast to epithermal flux ratio of 0.085 ± 0.002 at the patient position and produces 86.2 ± 0.2 kW fission power. The 41% increase in epithermal flux that heavy water provides is probably not great enough to justify the difficulty and expense of a heavy water coolant. With these epithermal fluxes at the patient position, a BNCT treatment would require 1-2 minutes and therefore would provide the Boston area with a unique facility capable of high patient throughput.

L.3 Thermal Hydraulics and Safety Analysis

The fission converter primary cooling system consists of a single loop with one pump, one heat exchanger, and associated valves and piping. The primary cooling loop model was implemented in the code TAMLOC [6]. Independent calculations have shown that flow instability will not occur because the power level is so low [69].

The objective of the safety analysis was to evaluate the ability of the facility to operate without hazard to the health and safety of the public. Many transients were analyzed as part of this study. The following transients were shown to cause nucleate boiling and hence should initiate a reactor scram within the allowed time. Nucleate Boiling in the fuel plates does not imply fuel damage will occur.

A Loss of flow without shutter closure transient causes occurrence of nucleate boiling after about 9 seconds. If the cadmium shutter closes during this time, nucleate boiling could be avoided, so the shutter activation mechanism should be designed so that the closure time is less than 9 seconds. In a loss of heat sink without shutter closure transient, nucleate boiling is reached after about 530 seconds.

LOCA without shutter closure was studied, since it is a more restrictive transient than LOCA with shutter closure. A break in the cold leg allowing a break flow rate of 12 gpm was chosen. After the pump trips on a low mass flow rate signal and coasts down, the fuel elements are cooled by natural circulation only. For this situation, nucleate boiling occurs 5.5 seconds from the beginning of the transient; at that time the clad temperature reaches the nucleate boiling temperature of 106°C. Although the inlet temperature decreases due to the loss of coolant mass, the outlet temperature increases, reaching the saturation temperature about 11 seconds into the transient.

Containment tank strength calculations show that an inner tank thickness of 18 mm of aluminum is required to minimize deflection and an outer tank thickness of 10 mm of aluminum is sufficient to contain the water if inner tank leakage occurs.

L.4 Conclusions

This study has shown that a fission converter-based epithermal neutron beam for a future BNCT facility at the MITR-II is feasible and that the design warrants further study.

Calculations have shown that a fission converter-based beam can provide an epithermal beam with high intensity and low fast neutron contamination superior to current beams. Also, thermal hydraulics simulations show that, from a reactor safety standpoint, such a facility can be operated in a safe manner. The cost of the facility was not analyzed by the group due to time constraints, however, a subsequent review by Michael McGuire provided an initial cost estimate of \$2.2 million, which includes the converter, filter, shutter, and patient treatment room. The fission converter-based epithermal beam design should be developed further and if clinical trials of BNCT are successful, a final design should be implemented.

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