UNIVERSITY OF FLORIDA
TRAINING REACTOR
LICENSE NO. R-56
DOCKET NO. 50-83

UPDATED
LICENSE RENEWAL APPLICATION
FOR
CONVERSION FROM HEU TO LEU FUEL

SAFETY ANALYSIS REPORT,
AND
TECHNICAL SPECIFICATIONS

REDACTED VERSION*

SECURITY-RELATED INFORMATION REMOVED

*REDACTED TEXT AND FIGURES BLACKED OUR OR DENOTED BY BRACKETS

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Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Re: Facility License R-56, Docket Number 50-83

Subject: Document Submittal for License Renewal for the University of Florida Training Reactor

The following documents are being submitted for license renewal:

Appendix 14.1 UFTR Updated Technical Specifications
Safety Assessment Report To Cover Analyses of UFTR Conversion from HEU to LEU Fuel

If further information is required, please let me know.

Sincerely,

Brian Shea

Brian Shea
Reactor Manager UFTR

DIANA L. DAMPIER
Notary Public - State of Florida
My Comm. Expires Jul 20, 2013
Commission # DD 894207
Bonded Through National Notary Assn.

Diana L. Dampier

Copies: Duane Hardesty, NRC Project Manager

Sworn and subscribed this 28 day of September, 2009

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APPENDIX 14.1

TECHNICAL SPECIFICATIONS

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TECHNICAL SPECIFICATIONS FOR THE UNIVERSITY OF FLORIDA TRAINING REACTOR

1.0 General

The University of Florida Training Reactor (UFTR) is operated by the Department of Nuclear and Radiological Engineering of the University of Florida. The UFTR is a non-power reactor used for instructional and research activities. The reactor is a modified Argonaut type, a light water and graphite moderated, graphite reflected, light water cooled reactor and operates at a nominal maximum steady state power level of 100 kWth.

1.1 DEFINITIONS*

Abnormal Occurrences: An abnormal occurrence is any one of the following:

(1) Operating the reactor with a safety system setting less conservative than specified in the Limiting Safety System Setting section of the Technical Specifications.

(2) Operating the reactor in violation of a limiting condition for operation.

(3) A malfunction of a safety system component or other component or system malfunction that could, or threatens to, render the system incapable of performing its intended safety function.

(4) A release of fission products from the reactor fuel of a magnitude to indicate a failure of the fuel cladding.

(5) An uncontrolled or unanticipated change in reactivity greater than one dollar (Reactor trips resulting from a known cause are excluded).

(6) An observed inadequacy in the implementation of either administrative or procedural controls such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor.

(7) An uncontrolled or unanticipated release of radioactivity to the environment.

* The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation

Blade-Drop Time: The blade-drop time is the elapsed time between the instant a limiting safety system set point is reached or a manual scram is initiated and the instant that the blade is fully inserted.

Certified Operator: An individual authorized by the Nuclear Regulatory Commission to carry out the duties and responsibilities associated with the position requiring the certification.

Channel Calibration: A channel calibration is an adjustment of the channel components such that its output responds, within specified range and accuracy, to known values of the parameter which the channel measures. Calibration shall encompass the entire channel, including readouts, alarms, or trips.

Channel Check: A channel check is a qualitative verification of acceptable performance by observation of channel behavior. This verification shall include comparison of the channel with other independent channels or methods of measuring the same variable.

Channel Test: A channel test is the introduction of an input signal into the channel to verify that it is operable.

Confinement: Confinement means a closure on the reactor room air volume such that the movement of air into it and out of the reactor room is through a controlled path.

Independent Experiment: An independent experiment is one that is not connected by a mechanical, chemical, or electrical link.

Inhibit: An inhibit is a device that prevents the withdrawal of control blades under a potentially unsafe condition.

Measured Value: The measured value of a parameter is the value as it appears at the output of a measuring channel.

Measuring Channel: The measuring channel is the combination of sensor, lines, amplifiers, and output devices that are connected for the purpose of measuring the value of a process variable.

Movable Experiment: A movable experiment is one where it is intended that the entire experiment may be moved in or near the core or into and out of the reactor while the reactor is operating, or having incore components during operation.

Nonsecured Experiment: A nonsecured experiment, where it is intended that the experiment should not move while the reactor is operating, is held in place with less restraint than a secured experiment.

Operable: A system or component is operable when it is capable of performing its intended function in a normal manner.

Operating: A system or component is operating when it is performing its intended function in a normal manner.

Reactor Operating: The reactor is considered to be operating whenever it is not secured or shutdown.

Reactor Operator (Class B Reactor Operator): Any individual who is certified to manipulate the controls of the reactor.

Reactor Safety System: The reactor safety system is that combination of measuring channels and associated circuitry that are designed to initiate automatic protective action or to provide information for initiation of manual protective action.

Reactor Secured: The reactor is secured when it contains insufficient fissile material or moderator present in the reactor, adjacent experiments or control blades, to attain criticality under optimum available conditions of moderation and reflection,

or

(1) the reactor is shutdown, (2) electrical power to the control blade circuits is switched off and the switch key is in proper custody, (3) no work is in progress involving core fuel, core structure, installed control blades or control blade drives unless they are physically decoupled from the control blades, and (4) no experiments are being moved or serviced that have, on movement, a reactivity worth exceeding the maximum value allowed for a single experiment or one dollar, whichever is smaller.

Reactor Startup: A reactor startup is a series of operator manipulations of reactor controls (in accordance with approved procedures) intended to bring the reactor to a k_{eff} of 0.99 or greater. It does not include control blade manipulations made for purposes of testing equipment or component operability within a k_{eff} of 0.99 or less.

Reactor Shutdown: The reactor is shut down when all control blades are inserted and the reactor is subcritical by a margin greater than $2\%\Delta k/k$. When calculating the subcritical margin, no credit shall be taken for experiments, temperature effects or xenon poisoning.

Reactor Trip: A reactor trip is considered to occur whenever one of the following two actions take place:

(1) Blade-Drop Trip — a gravity drop of all control blades into the reactor core as a result of terminating electrical power to the blade drive magnetic clutches.

(2) Full-Trip — the water is dumped from the reactor core by the safety actuation of the dump valve in addition to the blade-drop trip.

Reference Core Condition: The condition of the core when it is at ambient temperature (~20°C) and the reactivity worth of the xenon is zero (cold, clean of xenon and critical).

Reportable Occurrence: A reportable occurrence is any of the conditions described in Section 6.6.2 of this specification.

Research Reactor: A research reactor is a device designed to support a self-sustaining neutron chain reaction to supply neutrons or ionizing radiation for research, developmental, educational, training, or experimental purposes, and which may have provisions for the production of nonfissile radioisotopes.

Safety Channel: A safety channel is a measuring channel in the reactor safety system.

Secured Experiment: A secured experiment is a stationary experiment held firmly in place by a mechanical device secured to the reactor structure or by gravity, providing that the weight of the experiment is such that it cannot be moved by a force of less than 60 lb.

Secured Experiment with Movable Parts: A secured experiment with movable parts is one that contains parts that are intended to be moved while the reactor is operating.

Senior Reactor Operator (Class A Reactor Operator): Any individual who is certified to direct the activities of Reactor Operators (Class B reactor operators); such an individual is also a reactor operator.

Should, Shall, and May: The word "shall" is used to denote a requirement; the word "should" to denote a recommendation; and the word "may" to denote permission, neither a requirement nor a recommendation.

Shutdown Margin: Shutdown margin is the minimum shutdown reactivity necessary to provide confidence that the system can be made subcritical by means of the control and safety systems starting from any permissible operating condition and with the most reactive blade in the most reactive position and that the reactor will remain subcritical without further operator action.

Shutdown Reactivity: Shutdown reactivity is the value of the reactivity of the reactor with all control blades in their least reactive positions (e.g., all inserted). The value of the shutdown reactivity includes the reactivity value of all installed experiments and is determined with the reactor at ambient conditions.

Unscheduled Shutdown: An unscheduled shutdown is any unplanned shutdown of the reactor after startup has been initiated.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 Safety Limits

Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. The principal physical barrier shall be the fuel cladding.

Applicability: These specifications apply to the variables that affect thermal, hydraulic, and materials performance of the core.

Objective: To ensure fuel cladding integrity.

Specifications:

(1) The fuel and cladding temperatures shall not exceed 986°F.

Bases: Operating experience and detailed calculations of Argonaut reactors and for the HEU to LEU conversion have demonstrated that Specification (1) suffices to maintain the core flow conditions to assure no onset of nucleate boiling within the core and the fuel and fuel cladding below temperatures at which fuel degradation would occur.

2.2 Limiting Safety System Settings

Limiting safety system settings for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions.

Applicability: These specifications are applicable to the reactor safety system setpoints.

Objective: To ensure that automatic protective action is initiated before exceeding a safety limit or before creating a radioactive hazard that is not considered under safety limits.

Specifications: The limiting safety system settings shall be

(1) Power level at any flow rate shall not exceed 119 kWth.

(2) The primary coolant flow rate shall be:

  (a) greater than 36 gpm at all power levels greater than 1 watt if the fuel coolant channel spacing tolerance is $\leq$ 15 mils.

  (b) greater than 41 gpm at all power levels greater than 1 watt if the fuel coolant channel spacing tolerance is $\leq$ 20 mils.

(3) The average primary coolant

(a) Inlet temperature shall not exceed 109°F when the fuel coolant channel spacing tolerance is ≤ 10 mils.

(b) Inlet temperature shall not exceed 99°F when the fuel coolant channel spacing tolerance is ≤ 20 mils.

(c) Outlet temperature shall not exceed 155°F when measured at any fuel box outlet.

(4) The reactor period shall not be faster than 3 sec.

(5) The high voltage applied to Safety Channel 1 and Safety Channel 2 neutron detection chambers shall be 90% or more of the established normal value.

(6) The primary coolant flow rate at the return line shall be greater than zero.

(7) The primary coolant core level shall be at least 2 in. above the fuel.

(8) The secondary coolant flow shall satisfy one of the following two conditions when the reactor is being operated at power levels equal to or larger than 1 kW:

(a) Power shall be provided to the well pump <u>and</u> the well water flow rate shall be larger than 60 gpm when using the well system for secondary cooling;

<u>or</u>

(b) The water flow rate shall be larger than 8 gpm <u>when</u> using the city water system for secondary cooling.

(9) The dilution fan rpm indication shall be 95% or more of the established normal value.

(10) The water level in the shield tank shall not be reduced 6 in. below the established normal level.

<u>Bases</u>: The University of Florida Training Reactor (UFTR) limiting safety system settings (LSSS) are established from operating experience and safety considerations. LSSS 2.2 (1) through (8) are established for the protection of the fuel, the fuel cladding, and the reactor core integrity. The primary and secondary bulk coolant temperatures, as well as the outlet temperatures of all six fuel boxes, are monitored and recorded in the control room. LSSS 2.2 (9) is established for the protection of reactor personnel in relation to accumulation of Argon-41 in the reactor cell and for the control of radioactive gaseous effluents released from the cell. LSSS 2.2 (10) is established to protect reactor personnel from potential external radiation hazards caused by loss of biological shielding.

3.0 LIMITING CONDITIONS FOR OPERATION

Limiting conditions for operation are the lowest functional capabilities or performance levels required of equipment for safe operation of the facility.

3.1 Reactivity Limitations

Applicability: These specifications apply to the parameters which describe the reactivity condition of the core.

Objectives: To ensure that the reactor cannot achieve prompt criticality, that the fuel temperature does not reach melting point, that the reactor can be safely shutdown under any condition and to limit the reactivity insertion rate to levels commensurate with efficient and safe reactor operation.

Specifications: The reactor shall not be critical unless the following conditions exist:

(1) Shutdown Margin: The minimum shutdown margin, with the most reactive control blade fully withdrawn, shall not be less than 2% $\Delta k/k$.

(2) Excess Reactivity: The core excess reactivity at cold critical, without xenon poisoning, shall not exceed 1.4% $\Delta k/k$.

(3) Coefficients of Reactivity: The primary coolant void and temperature coefficients of reactivity shall be negative.

(4) Maximum Single Blade Reactivity Insertion Rate: The reactivity insertion rate for a single control blade shall not exceed 0.06% $\Delta k/k$/sec, when determined as an average over any 10 sec of blade travel time from the characteristic experimental integral blade reactivity worth curve.

(5) Experimental Limitations: The reactivity limitations associated with experiments are specified in Section 3.6 of these specifications.

Bases: Specification (1) ensures that a reactor shutdown can be established with the most reactive blade out of the core. Specification (2) is based on analysis documented in SAR Chapter 4, Section 4.1.2 and Chapter 13 to prevent the possibility that an inadvertent sudden excess of reactivity insertion could release significant energy to damage the fuel or cladding. Specification (3) is based on safe requirements for operation of light water reactors. Specification (4) limits the reactivity insertion rate to levels commensurate with efficient and safe reactor operation and Specification (5) is based on the reactor control system capabilities (20-sec positive period limitation). These limits are also established based on UFTR operating experience.

3.2 Reactor Control and Safety Systems

Applicability: These specifications apply to the reactor control and safety systems.

3.2.1 Reactor Control System

Objectives: To specify minimum acceptable capability and level of equipment for the reactor control system, range of reactivity insertion rate and interlocks to assure safe operation of the reactor.

Specifications:

(1) Four cadmium-tipped, semaphore-type blades shall be used for reactor control. The control blades shall be protected by shrouds to ensure freedom of motion.

(2) Only one control blade can be raised by the manual reactor controls at any one time. The safety blades shall not be used to raise reactor power simultaneously with the regulating blade when the reactor control system is in the automatic mode of operation.

(3) A reactor startup shall not be commenced unless the reactor control system is operable. The auto controller is not required to be operable if it is not needed for the operation involved.

(4) The control blade drop time shall not exceed 1.5 seconds from initiation of blade drop to full insertion (blade-drop time), as determined according to surveillance requirements.

(5) The following control blade withdrawal inhibit interlocks shall be operable for reactor operation:

(a) a source (startup) count rate of less than 2 cps (as measured by the wide range drawer operating on extended range).

(b) a reactor period less than 10 sec.

(c) safety channels 1 and 2 and wide range drawer calibration switches not in OPERATE condition.

(d) attempt to raise any two or more blades simultaneously when the reactor is in manual mode, or two or more safety blades simultaneously when the reactor is in automatic mode.

(e) power is raised in the automatic mode at a period faster than 30 sec. (The automatic controller action is to inhibit further regulating blade withdrawal or drive the regulating blade down until the period is ≥ 30 sec.)

(6) Following maintenance or modification to the reactor control system, an operability test and calibration of the affected portion of the system, including verification of control blade drive speed, shall be performed before the system is considered operable.

(7) The reactor shall be shut down when the main alternating current (ac) power is not operating.

(8) The primary coolant pump shall be energized during reactor operations.

Bases: The operator has available digital control blade position indicators for the three safety blades and the regulating blade. The three safety blades can only be manipulated by the UP-DOWN blade switches (manual); the regulating blade can be manually controlled or placed under automatic control, which uses the linear channel as the measuring channel, and a percent of power setting for control. Specifications (1) and (4) ensure that the reactor can be shut down promptly when a scram signal is initiated. Specification (2) ensures there is no possibility to reach a prompt critical condition and to limit the reactivity insertion rate to levels commensurate with efficient and safe reactor operation. Specification (3) ensures the reactor control system operability for startup and that the automatic controller is operable if needed. Specifications (5) (a), (b), (d) and (e) ensure that blade movement is performed under proper monitoring with assured source count rate and safe period either under manual or automatic control. Specification (5) (c) ensures that the operator is monitoring power changes during blade movement. Specification (6) ensures checking for proper functioning of the control blade system prior to reactor operation after maintenance has been performed on the system. Specification (7) ensures that the reactor is shut down during power outrages. And Specification (8) ensures that the reactor is shut down when the primary coolant pump is not energized.

3.2.2 Reactor Safety System

Objective: To ensure that sufficient information is available to the operator allowing safe operation of the reactor.

Specifications:

(1) The reactor shall not be started unless the reactor safety system is operable in accordance with Table 3-1.

(2) Tests for operability shall be made in accordance with Table 3-2.

Bases: Specification (1) ensures that no operation will be performed under abnormal conditions as listed in Table 3-1 and that the necessary reactor control system trip functions are operable in case of occurrence of any of these conditions. The two independent reactor safety channels provide redundant protection and information on reactor power in the range 1%-150% of full power. The linear power channel is the most accurate neutron instrumentation channel and also provides a signal for reactor control in automatic mode. The percent of power information is displayed by the linear channel two-pen recorder. It does not provide a protective function. The log wide range drawer provides a series of information, inhibit, and protection functions from extended source range to full power. The safety channel 1 signal and the period protection signal are derived from the wide range drawer. The wide range drawer provides protection during startup through the source count rate interlock (2 cps), 10-sec period inhibit and the 3-sec period trip. The primary and secondary coolant flow rate, temperature and level sensing instrumentation provides information and protection over the entire range of reactor operations and is proven to be conservative from a safety viewpoint. The key switch prevents unauthorized operation of the reactor and is an additional full trip (manual scram) control available to the operator. The core level trip provides redundant protection to the primary flow trip. The core level trip acts as an inhibit during startup until the minimum core water level is reached. As stated in Section 2, these limits were set based on operating experience and safety considerations.

Specification (2) ensures proper surveillance of the components of the reactor safety system and scram functions to assure operability prior to startup.

3.2.3 Reactor Control and Safety Systems Measuring Channels

Objective: To specify the minimum number and type of acceptable measuring channels for the reactor safety system and safety related instrumentation.

Specification: The minimum number and type of measuring channels operable and providing information to the control room operator required for reactor operation are presented in Table 3-3.

Bases: Table 3-3 specifies the minimum number of acceptable components for the reactor safety system and related instrumentation to assure the proper functioning of the reactor safety systems as specified in SAR Chapter 7.

The reactor control system provides the operator with reactivity control devices to control the reactor within the specified limits of reactivity insertion rate and power level. The operator has available digital blade position indicators for the three safety blades and the regulating blade. The three safety blades can only be manipulated by the UP-DOWN blade switches (manual); the regulating blade can be manually controlled or placed under automatic control, which uses the linear channel as the measuring channel, and a percent of power setting control. The two independent reactor safety channels provide redundant protection and information on reactor power in the range 1% - 150% of full power. The linear power channel is the most accurate neutron instrumentation channel, and provides a signal for reactor control in automatic mode. The percent of power information is displayed by the linear channel two-pen recorder. It does not provide a protective function. The log wide range drawer provides a series of information, inhibit, and protection function from extended source range to full power. The safety channel 1 signal and the period protection signal are derived from the wide range drawer. The wide range drawer provides protection during startup through the source count rate interlock (2 cps), 10 sec period inhibit, and the 3 sec period trip. The primary and secondary coolant flow rate, temperature and level sensing instrumentation provides information and protection over the entire range of reactor operations and is proven to be conservative from a safety viewpoint. The key switch prevents unauthorized operation of the reactor and is an additional full trip (manual scram) control available to the operator. The core level trip provides redundant protection to the primary flow trip. The core level trip acts as an inhibit during startup until the minimum core water level is reached.

3.3 Primary Water Quality

Applicability: These specifications apply to the reactor cooling system and water in contact with fuel plates or elements.

Objective: To minimize corrosion of the aluminum cladding of fuel plates and activation of dissolved materials.

Specifications:

(1) Primary water temperature shall not exceed 155°F.

(2) Primary water shall be demineralized, light water with a specific resistivity of not less than 0.5 megohm-cm after the reactor is operated for more than 6 hours.

(3) Primary water shall be sampled and evaporatively concentrated, and the gross radioactivity of the residue shall be measured with an adequate measuring channel. This specification procedure shall prevail (a) during the weekly checkout, (b) upon the appearance of any unusual radioactivity in the primary water or the primary water demineralizers, and (c) before the release of any primary water from the site.

(4) The primary equipment pit water level sensor shall alarm in the control room whenever a detectable amount of water (1 in. above floor level) exists in the equipment pit.

(5) Primary water pH shall be <7.0,

Bases: Specifications 3.3 (1), 3.3 (2) and 3.3 (5) are designed to protect fuel element cladding integrity and are based upon operating experience. At the specified reactivity, the activation products (of trace minerals) do not exceed acceptable limits. Specification 3.3 (3) is designed to detect and identify fission products resulting from fuel failure and to fulfill reportability requirements pertaining to liquid wastes. Specification 3.3 (4) is designed to alert the operator to potential loss of primary coolant, to prevent reactor operations with a reduced water inventory and to minimize the possibility of an uncontrolled release of primary coolant to the environs.

3.4 Reactor Vent System

Applicability: These specifications apply to the equipment required for controlled release of gaseous radioactive effluent to the environment via the stack or its confinement within the reactor cell.

Objective: To limit the amount and concentration of radioactivity in the effluent from the reactor cell and reduce the back leakage of radioactivity into the reactor cell under normal operations and from the cell under emergency conditions.

Specifications:

(1) The reactor vent system shall be operated at all times during reactor operation. In addition, the vent system shall be operated until the stack monitor indicates less than 10 counts per-second (cps) unless otherwise-indicated by facility conditions to include loss of building electrical power, equipment failure or maintenance, cycling console power to dump primary coolant or to conduct tests and surveillances and initiating the evacuation alarm for tests and surveillances including emergency drills and demonstrations. The reactor vent system shall be immediately secured upon detection of: a failure in the monitoring system, a failure of the absolute filter, or an unanticipated high stack count rate.

(2) The reactor vent system shall be capable of maintaining an air flow rate between 1 and 400 cfm from the reactor cavity whenever the reactor is operating and as specified in these Technical Specifications.

(3) The diluting fan shall be operated whenever the reactor is operating and as otherwise specified in these Technical Specifications, at an exhaust flow rate larger than 10,000 cfm.

(4) The air conditioning/ventilation system and reactor vent system are automatically shut off whenever the reactor building evacuation alarm is automatically or manually actuated.

(5) All doors to the reactor cell shall normally be closed while the reactor is operating. Transit is not prohibited through the exit chamber and control room doors.

(6) The reactor vent system shall have a backup means for quantifying the radioactivity in the effluent during abnormal or emergency operating conditions

where venting could be used to reduce cell radionuclide concentrations for ALARA considerations.

Bases: Under normal conditions, to affect controlled release of gaseous activity through the reactor vent system, a negative cell pressure is required so that any building leakage will be inward. Under normal shutdown conditions with significant Argon-41 inventory in the reactor cavity, operation of the core vent system prevents unnecessary exposure from gas leakage back into the cell. Under emergency conditions, the reactor vent system will be shut down and the damper closed, thus minimizing leakage of radioactivity from the reactor cell unless venting is required.

3.5 Radiation Monitoring Systems and Radioactive Effluents

Applicability: These specifications apply to the radiation monitoring systems and to the limits on radioactive effluents.

Objective: To specify the minimum equipment or the lowest acceptable level of performance for the radiation monitoring systems and limits for effluents.

Specifications: The reactor shall not be operated unless the conditions presented in Sections 3.5.1 through 3.5.6 are met.

3.5.1 Area Radiation Monitors

The reactor cell shall be monitored by at least three area radiation monitors, two of which shall be capable of audibly warning personnel of high radiation levels. The output of at least two of the monitors shall be indicated and recorded in the control room. The number required and setpoints for the radiation monitors shall be in accordance with Table 3-4 including more conservative setpoints if desired.

3.5.2 Argon-41 Discharge

The following operational limits are specified for the discharge of Argon-41 to the environment:

(1) The concentration of Argon-41 in the gaseous effluent discharge of the UFTR is determined by averaging it over a consecutive 30-day period.

(2) The dilution resulting from the operation of the stack dilution fan (flow rate of 10,000 cfm or more) and atmospheric dilution of the stack plume (a factor of 200) may be taken into account when calculating this concentration.

(3) When calculated as above, the discharge concentration of Argon-41 shall not exceed 1.0×10^{-8} μCi/ml. Operation of the UFTR shall be such that this maximum concentration (averaged over a month) is not exceeded.

3.5.3 Reactor Vent/Stack Monitoring System

(1) Whenever the reactor vent system is operating, air drawn through the reactor vent system shall be continuously monitored for gross count rate of radioactive gases. The output of the monitor shall be indicated and recorded in the control room. Operable functions and alarm settings shall be as delineated in Table 3-4.

(2) Whenever venting is to be used to reduce cell radionuclide concentrations during abnormal or emergency conditions, then the radioactivity in the effluent shall be quantified prior to initiating controlled venting.

(3) Whenever significant changes are noted, the reactor air cavity flow may be periodically analyzed to minimize Argon-41 releases to the environment while maintaining a negative pressure within the reactor cavity to minimize potential radioactive hazards to reactor personnel.

3.5.4 Air Particulate Monitor

The reactor cell environment shall be monitored by at least one air particulate monitor, capable of audibly warning personnel of radioactive particulate airborne contamination in the cell atmosphere.

3.5.5 Liquid Effluents Discharge

(1) The liquid effluent from the aboveground holdup tank shall be sampled and the radioactivity measured before release to the sanitary sewage system which is allowed in conformance with 10 CFR 20.1301.

(2) Releases of radioactive effluents from the external waste water holdup tank shall be in compliance with the limits specified in 10 CFR 20, Appendix B, Table 2, Column 2, as specified in 10 CFR 20.1302.

3.5.6 Solid Radioactive Waste Disposal

Solid radioactive waste disposal shall be accomplished in compliance with applicable regulations and under the control of the Radiation Control Office of the University of Florida.

3.5.7 Bases

The area radiation monitoring system, stack monitoring system and air particulate detector(s) provide information to the operator indicating radiation and airborne contamination levels under the full range of operating conditions. Audible indicators and alarm lights indicate (via monitored parameters) when corrective operator action is required, and (in the case of the area radiation monitors) a warning light indicates situations recommending or requiring special operator attention and evaluation. Argon-41 discharges are limited to a monthly average which is less than the effluent concentration limit in 10 CFR 20, Appendix B, Table 2, and liquid and solid radioactive wastes are regulated and controlled to assure compliance with legal requirements.

3.6 Limitations on Experiments

Applicability: These specifications apply to all experiments or experimental devices installed in the reactor core or its experimental facilities.

Objectives: The objectives are to assure operational safety and prevent damage to the reactor facility, reactor fuel, reactor core, and associated equipment; to prevent exceeding the reactor safety limits; and to minimize potential personnel and equipment hazards from experimental devices.

Specifications:

(1) General

The reactor manager and the radiation control officer (or their duly appointed representatives) shall review and approve in writing all proposed experiments prior to their performance. The reactor manager shall refer to the Reactor Safety Review Subcommittee (RSRS) the evaluation of the safety aspects of new experiments and all changes to the facility that may be necessitated by the requirements of experiments that may have safety significance. When experiments contain hazardous materials or substances which, upon irradiation in the reactor, can be converted into a material with significant potential hazards, a determination will be made about the acceptable reactor power level and length of irradiation, taking into account such factors as: isotope identity and chemical and physical form and containment; toxicity; potential for contamination of the facility or the environment; problems in removal or handling after irradiation including containment, transfer, and eventual disposition. Guidance should be obtained from the ANSI/ANS 15.1. Experimental apparatus, material, or equipment to be inserted in the reactor shall be reviewed to ensure compatibility with the safe operation of the reactor.

(2) Classification of Experiments

Class I — Routine experiments, such as gold foil irradiation. This class shall be approved by the reactor manager; the radiation control officer may be informed if deemed necessary.

Class II— Relatively routine experiments that need to be documented for each new group of experimenters performing them, or whenever the experiment has not been carried out for one calendar year or more by the original experimenter, and that pose no hazard to the reactor, the personnel, or the public. This class shall be approved by the reactor manager and the radiation control officer.

Class III— Experiments that pose significant questions regarding the safety of the reactor, personnel, or the public. This class shall be approved by the reactor manager and the radiation control officer, after review and approval by the Reactor Safety Review Subcommittee (RSRS).

Class IV— Experiments that have a significant potential for hazard to the reactor, the personnel, or the public. This class shall be approved by the reactor manager and radiation control officer after review and approval by the RSRS and specific emergency operating instructions shall be established for conducting the experiments.

(3) Reactivity Limitations on Experiments

(a) The absolute reactivity worth of any single movable or nonsecured experiment shall not exceed 0:6% $\Delta k/k$.

(b) The total absolute reactivity worth of all experiments shall not exceed 1.4% $\Delta k/k$.

(c) When determining the absolute reactivity worth of an experiment, no credit shall be taken for temperature effects.

(d) An experiment shall not be inserted or removed unless all the control blades are fully inserted or its absolute reactivity worth is known to be less than that which could cause a positive 20-sec stable period.

(4) Explosive Materials

Explosive materials shall not be irradiated in the core unless the irradiation container has the ability to contain 200% of the potential energy to be released if the explosive is detonated. Following an adequate safety analysis, explosive materials may be irradiated using beams extracted from the core as well.

(5) Thermal-Hydraulic Effects

Experiments shall be designed so that during normal operation, or failure, the thermal hydraulic parameters of the core do not exceed the safety limits.

(6) Chemical Effects

Experiments shall be designed so that during normal operation, or failure, the physical barrier described in Section 2.1 will not be compromised by either chemical or blast effects from the experiment.

(7) Fueled Experiments

A limit should be established on the inventory of fission products in any experiment containing fissile material, according to its potential hazard and as determined by the RSRS.

(8) Radioactive Releases from Experiments

Class III and Class IV experiments shall be evaluated for their potential release of airborne radioactivity and limits shall be established for the permissible concentration of radioisotopes in the experiments, according to the 10 CFR 20 limitations for exposure of individuals in restricted and unrestricted areas.

Bases: These eight (8) specifications generally ensure that an adequate review process is followed to assure the safe operation, proper conditions, and adherence to procedures for all experiments. The classification of experiments clearly delineates the responsibility for approving experiments according to their potential hazards, to ensure that potentially hazardous experiments are analyzed for their safety implications, and that appropriate procedures are established for their execution. The reactivity limitations on experiments are established to prevent prompt criticality by limiting the worth of movable or nonsecured experiments, to prevent a reactivity insertion larger than the stipulated maximum step reactivity insertion in the accident analysis, and to allow for reactivity control of experiments within the reactor control system capabilities (20-sec positive period limitation). These specifications limit the irradiation of explosive materials. Explosive materials are defined as those materials normally used to produce explosive or detonating effects, materials that can chemically combine to produce explosions or detonations, or any materials that can undergo explosive decomposition under influence of neutron, gamma, or heat flux of the reactor or as defined by applicable standards. These specifications also limit the amount of fissile materials that can be irradiated in the reactor according to its potential hazard and the reactor system's capability to handle a potential release to the cell environment.

3.7 Reactor Building Evacuation Alarm

Applicability: These specifications apply to the systems and equipment required for the evacuation of the reactor cell and the reactor building (including the reactor annex).

Objective: To specify conditions to actuate the evacuation alarm.

Specifications: The reactor cell and the reactor building shall be evacuated when any of the following conditions exist:

(1) The evacuation alarm is actuated automatically when two area radiation monitors alarm high ($\geq$25 mrem/hr) in coincidence.

(2) The evacuation alarm is actuated manually when an air particulate monitor is in a valid alarm condition.

(3) The evacuation alarm is actuated manually when a reactor operator detects a potentially hazardous radiological condition and preventive actions are required to protect the health and safety of operating personnel and the general public.

Bases: To provide early and orderly evacuation of the reactor cell and the reactor building and to minimize radioactive hazards to the operating personnel and reactor building occupants. No response is required for trip tests and demonstrations of the evacuation alarm.

3.8 Fuel and Fuel Handling

Applicability: These specifications apply to the arrangement of fuel elements in core and in storage, as well as the handling of fuel elements.

Objectives: The objectives are to establish the maximum core loading for reactivity control purposes, to establish proper fuel storage conditions and to establish fuel performance and fuel handling specifications with regard to radiological safety considerations.

Specifications:

(1) The maximum core fuel loading shall consist of 24 full fuel elements consisting of 14 plates each containing enriched uranium and clad with high purity aluminum.

(2) Fuel element loading and distribution in the core shall comply with approved fuel-handling procedures.

(3) Fuel elements exhibiting release of fission products because of cladding rupture shall, upon positive identification, be removed from the core. Significant fission product contamination of the primary water shall be treated as evidence of fuel element failure.

(4) The reactor shall not be operated if there is evidence of fuel element failure.

(5) All fuel shall be moved, handled and stored in accordance with approved procedures.

(6) Fuel elements or fueled devices shall be stored and handled out of core in a geometry such that the k_{eff} is less than 0.8 under optimum conditions of moderation and reflection.

(7) Irradiated fuel elements or fueled devices shall be stored so that temperatures do not exceed design values.

Bases: The core fuel loading is based on the present fuel configuration. The reactor systems do not have adequate engineering safeguards to continue operating with a detectable release of fission products into the primary coolant. The fuel is to be stored in a safe configuration and shall be handled according to approved written procedures for radiological safety purposes and adherence to limiting personnel radiation doses to as low as reasonably achievable (ALARA).

3.9 Radiological Environmental Monitoring Program

Applicability: This specification applies to the environmental radioactivity surveillances and surveys conducted by UFTR personnel and Radiation Control and Radiological Services Department personnel.

Objectives: The UFTR Radiological Environmental Monitoring Program is conducted to ensure that the radiological environmental impact of reactor operations is as low as reasonably achievable (ALARA); it is conducted in addition to the radiation monitoring and effluents control specified under Section 3.5 of these Technical Specifications.

Specifications: The Radiological Environmental Monitoring Program shall be conducted as specified below and under the supervision of the Radiation Control Officer.

(1) Monthly environmental radiation dose surveillance outside the restricted area shall be conducted by measuring the gamma doses at selected fixed locations surrounding the UFTR complex with acceptable personnel monitoring devices. A minimum of six independent locations shall be monitored. A review of potential causes shall be conducted whenever a measured dose of over 40 mrem/month at two or more locations is determined and a report shall be submitted to the RSRS for review.

(2) Radioactivity surveillance of the restricted area (reactor cell) shall be conducted as follows:

(a) Surface contamination in the restricted area shall be measured by taking random swipes in the reactor cell during the weekly checkout. Measured surface contamination greater than 100 dpm/100cm^2 beta-gamma or greater than 50 dpm/100 cm^2 alpha are limiting conditions for operation requiring review and possible radiological safety control actions.

(b) Airborne particulate contamination shall be measured using a high volume air sampler during the weekly checkout. Measured radioactive airborne contamination 25% above mean normal levels are limiting conditions for operation requiring review and possible radiological safety control actions.

(3) The following radiation surveys, using portable radiation monitors, are limiting conditions for operation:

(a) Surveys measuring radiation dose rates in the restricted area shall be conducted quarterly, at intervals not to exceed 4 months, and at any time a change in the normal radiation levels is noticed or expected. Radiation exposures shall be maintained within 10 CFR 20 limits for radiation workers.

(b) Surveys measuring the radiation dose rates in the unrestricted areas surrounding the UFTR complex shall be conducted quarterly, at intervals not to exceed 4 months, and at any time a change in the normal radiation levels is noticed or expected. Dose rates shall be within 10 CFR 20 limits for the general public.

Bases: The bases for establishing the Radiological Environmental Surveillance Program are the established limits for internal and external radiation exposure and requirements that radiation doses be maintained ALARA and the necessity to confirm and document that UFTR operations are conducted to be within the established limits.

Table 3-1 Specification for Reactor Safety System Trips

Specification	Type of safety system trip
<u>Automatic Trips</u>	
Period less than 3 sec	Full
Power at 119% of full power	Full
Loss of chamber high voltage (≥ 10%)	Full
Primary cooling system	Blade-drop
Loss of pump power Low water level in core (<42.5") No outlet flow Low inlet water flow (< 36 gpm for fuel coolant channel spacing tolerance at ≤ 15 mils); (< 41 gpm for fuel coolant channel spacing tolerance at ≤ 20 mils)	
Secondary cooling system (at power levels above 1 kW)	Blade-drop
Loss of flow (well water < 60 gpm, city water < 8 gpm) Loss of pump power	
High primary coolant average inlet temperature (≥ 109°F for fuel coolant channel spacing tolerance at ≤ 10 mils); (≥ 99°F for fuel coolant channel spacing tolerance at ≤ 20 mils)	Blade-drop
High primary coolant average outlet temperature (≥ 155°F)	Blade-drop
Shield Tank Low water level (6" below established normal level)	Blade-drop
Ventilation system	Blade-drop
Dilution fan RPM < 95% established normal level	
<u>Manual Trips</u>	
Manual scram bar	Blade-drop
Console key-switch OFF (two blades off bottom)	Full

Table 3-2 Safety System Operability Tests

Component or Scram Function	Frequency
Log-N period channel Power level safety channels	Before each reactor startup following a shutdown in excess of 6 hr, <u>and</u> after repair <u>or</u> deenergization caused by a power outage.
10% reduction of safety channels high voltage	4/year (4-month maximum interval)
Loss of primary coolant pump power	4/year (4-month maximum interval)
Loss of primary coolant level	4/year (4-month maximum interval)
Loss of primary coolant flow	4/year (4-month maximum interval)
High average primary coolant inlet temperature	With daily checkout
High average primary coolant fuel box outlet temperature	With daily checkout
Loss of secondary coolant flow (at power levels above 1 kW)	With daily checkout
Loss of secondary coolant well pump power	4/year (4-month maximum interval)
Loss of shield tank water level	4/year (4-month maximum interval)
Loss of dilution fan RPM	4/year (4-month maximum interval)
Manual scram bar	With daily checkout

Table 3-3 Minimum Number and Type of Measuring Channels Operable

Channel	No. operable
Safety 1 and 2 power channel	2
Linear Channel (with auto controller as appropriate)	1
Log N and period channel*	1
Startup channel*	1
Blade position indicator	4
Coolant flow indicator	1
Coolant temperature indicator	
Primary	7
Secondary	1
Core level	1
Ventilation system	
Core vent annunciator	1
Dilute fan annunciator	1
Dilute fan rpm	1

*Subsystems of the wide range drawer

Table 3-4 Radiation Monitoring System Settings

Type	No. of Required Operable Functions	Alarm(s) Setting	Purpose
Area Radiation Monitors	3 detecting 2 audioalarming 2 recording	5 mr/hr low level 25 mr/hr high level	Detect/alarm/record low and high level external radiation
Air Particulate Monitors	1 detecting 1 audioalarming 1 recording	Range adjusted according to APD* type (according to monitoring requirements)	Detect/alarm/record airborne radioactivity in the reactor cell
Stack Radiation Monitor	1 detecting 1 audioalarming 1 recording	(1) Fixed alarm at 4000 cps (2) Adjustable alarm per power level	Detect/alarm/record release of gaseous radioactive effluents in the reactor vent duct to the environs

*Air Particulate Detector

4.0 SURVEILLANCE REQUIREMENTS

Surveillance requirements relate to testing, calibration, or inspection to ensure that the necessary quality of systems and components is maintained; that facility operation will be within safety limits; and that the limiting conditions for operation will be met. Tests not performed within the specified frequency because of physical or administrative limitations including equipment failure and maintenance activities shall be performed before resuming normal operations.

4.1 Surveillance Pertaining to Safety Limits and Limiting Safety System Settings.

(1) Whenever an unscheduled shutdown occurs, an evaluation shall be conducted to determine whether a safety limit was exceeded.

(2) Safety system operability tests shall be performed in accordance with Table 3-2.

4.2 Surveillance Pertaining to Limiting Conditions for Operation

4.2.1 Reactivity Surveillance

Applicability: These specifications apply to the surveillance activities required for reactivity parameters.

Objective: To specify the frequency and type of testing to assure that reactor core parameters conform to specifications in Section 3.1.

Specifications:

(1) The reactivity worth and reactivity insertion rate of each control blade, the shutdown margin and excess reactivity shall be measured annually (at intervals not to exceed 14 months) or whenever physical or operational changes create a condition requiring reevaluation of core physics parameters.

(2) The temperature coefficient of reactivity shall be measured annually at intervals not to exceed 15 months.

(3) The void coefficient of reactivity shall be checked biennially to ensure that it is negative, at intervals not to exceed 30 months.

Bases: The measurements specified are sufficient to provide assurance that the reactor core parameters are maintained within the limits specified in Section 3.1.

4.2.2 Reactor Control and Safety System Surveillance

Applicability: These specifications apply to the surveillance activities required for the reactor control and safety systems.

Objective: To specify the frequency and type of testing or calibration to assure that reactor control and safety system operating parameters conform to specifications in Section 3.2.

Specifications:

(1) Control blade drop times, from the fully withdrawn position, shall be measured semiannually at intervals not to exceed 8 months. If maintenance is performed on a blade, the drive mechanism, or associated electronics, the blade-drop time shall be measured before the system is considered operable.

(2) The control blade full withdrawal and controlled insertion times shall be measured semiannually at intervals not to exceed 8 months.

(3) Tests, limits, and frequencies of tests for the control blade withdrawal inhibit interlocks operability tests shall be performed as listed in Table 4-1.

(4) The mechanical integrity of the control blades and drive system shall be inspected during each incore inspection but shall be fully checked at least once every 10 years at intervals not to exceed 12 years.

(5) Following maintenance or modification to the control blade system, an operability test and calibration of the affected portion of the system, including verification of control blade drive speed, shall be performed before the system is to be considered operable.

(6) The reactor shall not be started unless (a) the weekly checkout has been satisfactorily completed within 7 days prior to startup, (b) a daily checkout is satisfactorily completed within 8 hr prior to startup, and (c) no known condition exists that would prevent successful completion weekly or daily check.

(7) The limitations established under Paragraph 4.2.2 (6) (a) and (b) can be deleted if a reactor startup is made within 6 hr of a normal reactor shutdown on any one calendar day.

(8) The following channels shall be calibrated annually, at intervals not to exceed 15 months, and any time a significant change in channel performance is noted:

  (a) log N - period channel
  (b) power level safety channels (2)
  (c) linear power level channel

(d) primary coolant flow measuring system
(e) primary coolant temperature measuring system

(9) Following maintenance or modification to the reactor safety system, a channel test and calibration of the affected channel shall be performed before the reactor safety system is considered operable.

Bases: The frequency and type of test or calibration are defined based on operating experience and/or in accordance with ANSI/ANS-15.1-1990 to assure proper functioning of the systems and equipment that comprise the reactor control and safety systems.

4.2.3 Primary and Secondary Water Quality Surveillance

Applicability: These specifications apply to the surveillance activities required for the reactor coolant system.

Objective: To specify the frequency and type of testing or calibration to assure the reactor coolant system conforms to the specifications presented in Section 3.3

(1) The primary water resistivity shall be determined as follows:

(a) Primary water resistivity shall be measured during the weekly checkout by a portable conductivity meter using approved procedures. The measured value shall be larger than 0.4 megohm-cm.

(b) Primary water resistivity shall be measured during the daily checkout at both the inlet and outlet of the demineralizers (DM). The measured value, determined by an online conductivity meter annunciating in the control room, shall be larger than 0.5. megohm-cm at the outlet of the DM.

(2) Primary water shall be sampled and evaporatively concentrated, and the gross radioactivity of-the residue shall be measured with an adequate measuring channel. This specification procedure shall prevail.

(a) during the weekly checkout,

(b) upon the appearance of any unusual radioactivity in the primary water or the primary water demineralizers, and

(c) before the release of any primary water from the site.

(3) The primary water radioactivity shall be measured during the weekly checkout for gross β-γ and gross α activity.

(a) The measured α activity shall not exceed 50 dpm above background level.

(b) The measured β-γ activity shall not exceed 25% above mean normal activity level.

(4) The secondary water system shall be tested for radioactive contamination during the weekly checkout according to written procedures.

(5) The primary water pH value shall be measured during the weekly preoperational checkout using approved procedures. The measured value shall be < 7.0.

Bases: These specifications assure that necessary limits are maintained on fission products and other activated materials in primary and secondary coolant samples to provide assurance that the facility is operating in a safe and effective manner. The frequency and type of monitoring is based on operating experience.

4.2.4 Reactor Vent System Surveillance

Applicability: These specifications apply to the surveillance requirements for the reactor vent system.

Objective: To specify the frequency and type of testing to assure the reactor vent system conforms to the specifications presented in Section 3.4.

Specifications:

(1) The reactor vent system flow rates shall be measured annually at intervals not to exceed 15 months, as follows:

(a) reactor cavity exhaust duct flow rate (1 cfm < flow rate < 400 cfm);

(b) stack flow rate > 10,000 cfm.

(2) The following interlocks shall be tested as part of the weekly checkout:

(a) core vent system damper closed if diluting fan is not operating;

(b) reactor vent system shut off when the evacuation alarm is actuated.

Bases: These specifications assure the reactor vent system is operating as specified. The frequency and type of monitoring is based on operating experience and ANSI/ANS-15.1-1990.

4.2.5 Radiation Monitoring Systems and Radioactive Effluents Surveillance

Applicability: These specifications apply to the surveillance activities required for the radiation monitoring system and effluents released from the facility.

Objective: To specify frequency and type of testing to assure that the radiation monitoring system and effluent releases conform to the specifications in Section 3.5.

Specifications:

(1) The area radiation monitor channels, the stack monitor, and the air particulate monitor shall be verified to be operable before each reactor startup as required by the daily checkout. Calibration of radiation monitoring channels shall be performed quarterly at intervals not to exceed 4 months. Note: Portable radiation survey meters are not normally considered radiation monitoring channels, so there is no need for them to be calibrated quarterly unless used in place of an installed monitor.

(2) The Ar-41 concentration in the stack effluent shall be measured semiannually at intervals not to exceed 8 months.

(3) Releases of liquid effluents from the aboveground waste water holdup tank shall be sampled and the radioactivity measured before release to the sanitary sewage system which is allowed in conformance with 10 CFR 20 regulations.

(4) The reactor shall be placed in a reactor shutdown condition whenever Specification 4.2.5 (1) is not met.

(5) The reactor vent system shall be immediately secured upon detection of failure of the stack monitoring system.

Bases: Specification (1) assures the monitors are operable. Specification (2) provides the basis for limiting energy generation to assure Ar-41 releases are in accordance with 10 CFR 20, Appendix B, Table 2. Specification (3) ensures compliance with 10 CFR 20 for liquid releases from the site. Specifications (4) and (5) ensure that all releases of radioactivity will be controlled and monitored.

4.2.6 Surveillance of Experimental Limits

Applicability: This specification applies to the surveillance requirements for experiments installed in the UFTR core.

Objective: To prevent the conduct of experiments or irradiations which could damage the reactor or release an excessive amount of radioactivity.

Specifications:

(1) Surveillance to ensure that experiments meet the requirements of Section 3.6 shall be conducted before inserting each experiment into the reactor.

(2) The reactivity worth of an experiment shall be determined at approximately 1 W power level or as appropriate within limiting conditions for operation, before continuing reactor operation with the experiment.

Bases: Measurements of the reactivity worth of an experiment shall verify that the experiment is within the authorized reactivity limits.

4.2.7 Reactor Building Evacuation Alarm Surveillance

Applicability: These specifications apply to the surveillance requirements for the reactor building evacuation alarm.

Objectives: To assure that building alarm actuation, building occupants and reactor staff are responding as expected.

Specifications:

(1) The automatic actuation of the building evacuation alarm in coincidence with actuation of the high level alarm on two area monitors and the manual actuation of the evacuation alarm shall be tested as part of the weekly checkout.

(2) The automatic shutoff of the air handling system and the reactor vent system in coincidence with the building evacuation alarm shall be tested as part of the weekly checkout.

(3) Evacuation drills for facility personnel shall be conducted semiannually at intervals not to exceed 8 months.

Bases: Specification (1) ensures that the actuation of the building evacuation alarm is operable to alert occupants to the need to evacuate. Specification (2) ensures that the system responds correctly to a known input to assure isolation of the cell atmosphere upon actuation of the evacuation alarm. Specification (3) ensures that facility personnel and building occupants are familiar with emergency response procedures.

4.2.8 Surveillance Pertaining to Fuel

Applicability: These specifications apply to fuel installed in the core.

Objective: To verify integrity of the fuel.

Specifications:

(1) The incore reactor fuel elements shall be inspected every 10 years at intervals not to exceed 12 years, in a randomly chosen pattern, as deemed necessary. At least 8 elements will be inspected. At least 3 days shall have passed since the last operation at power ($\geq$ 1kW) before commencement of fuel handling to limit the possible/potential consequences of fuel handling accidents.

(2) Fuel-handling tools and procedures shall be reviewed for adequacy before fuel handling operations. The assignment of responsibilities and training of the fuel-handling crew shall be performed according to written procedures.

Bases: Specification (1) ensures the integrity of the fuel and Specification (2) assures that reactor and support staff are properly qualified to perform fuel handling and related activities.

Table 4-1 Control Blade Withdrawal Inhibit Interlocks Operability Tests

Inhibit	Limit	Frequency
Reactor Period	≤ 10 sec	Daily Checkout
Safety Channels and Wide Range Drawer not in OPERATE position	-	Daily Checkout
Multiple blade withdrawal	Any 2 or more blades simultaneously in Manual Any 2 safety blades in Automatic	Daily Checkout
Source count rate	< 2 cps	Verification only when count rate < 2cps during daily checkout

5.0 DESIGN FEATURES

Design features are specified to ensure that items important to safety are not changed without appropriate review. The items of concern are design features and parameters that were considered as limiting values (or significant for the protection of the reactor personnel and the general public) for the purpose of establishing safety limits, limiting safety system settings, or limiting conditions for operation.

5.1 Site

The UFTR is located on the University of Florida campus, at Gainesville, Florida, in the immediate vicinity of the buildings housing the College of Engineering and the College of Journalism. The Nuclear Science Center, which houses the Department of Nuclear and Radiological Engineering, is annexed to the reactor building.

The reactor shall be housed in a reinforced concrete cell in the reactor building. The reactor building is a "vault-type" building as defined in 10 CFR 73.2(o). The reactor building is divided into two distinct parts based upon the difference in utilization and their structure. The overall reactor building measures approximately 60 ft by 80 ft inside. The reactor cell area is 30 ft by 60 ft with 29 ft of head room, located at the north end of the building. The rest of the building is used for research laboratories, faculty offices, and graduate study areas.

5.2 Reactor Cell

The reactor shall be housed in a reinforced concrete cell in the reactor building. The reactor building is a "vault-type" building as defined in 10 CFR 73.2(o). The reactor building is divided into two distinct parts based upon the difference in utilization and structure. The overall reactor building measures approximately 60 ft by 80 ft inside. The reactor cell area is 30 ft by 60 ft with 29 ft of head room, located at the north end of the building. The rest of the building is used for research and instructional laboratories, faculty offices, and graduate study areas.

The reactor cell shall have an independent ventilation and air-conditioning system. The reactor vent effluents shall be discharged through the reactor stack about 30 ft above ground level.

All gases that may cause a hazard through neutron activation shall be exhausted from the reactor cell, reactor cavity, experiments or experimental facilities installed in or adjacent to the core or surrounding graphite and discharged to the environment through the reactor vent system and appropriately monitored for radioactivity, as specified under Chapter 3 of these Technical Specifications.

The 3-ton bridge crane shall not be used during reactor operation in a manner that could damage the control system and prevent it from performing its intended function. No load above 500 lb shall be lifted over the control blade drive units unless the control blades are fully inserted. The crane shall be operated during reactor operations only by a licensed reactor operator.

The following doors penetrate the reactor cell: (1) an exit chamber passageway from the cell to the UFTR building lower hallway, (2) a door from the control room to the UFTR building lower hallway, and (3) a freight door (10 ft x 12 ft) leading to the environs. A panel in the freight door serves as an emergency personnel exit from the reactor cell. The freight door and panel shall be locked to prevent entrance during reactor operation. The freight door and panel shall not be used for general access to or egress from the reactor cell. This is not meant to preclude use of these doors in connection with authorized activities when the reactor is not in operation.

5.3 Reactor Fuel

Fuel elements shall be of the general MTR type, with thin fuel plates clad with aluminum and containing uranium fuel enriched to no more than about 19.75% U-235. The fuel matrix may be fabricated from uranium silicide-aluminum (U_3Si_2-Al) using the powder metallurgy process. There shall be nominally 12.5 g U-235 per fuel plate.

The UFTR facility license authorizes the receipt, possession, and use of:

(1) up to 5.2 kg of contained uranium-235;
(2) a 1-Ci sealed plutonium-beryllium neutron source;
(3) an up-to-25-Ci antimony-beryllium neutron source.

Other neutron and gamma sources may be used if their use does not constitute an unreviewed safety question pursuant to 10 CFR 50.59 and if the sources meet the criteria established by the Technical Specifications.

5.4 Reactor Core

The core shall contain up to 24 fuel assemblies of 14 plates each. Up to six of these assemblies may be replaced with pairs of partial assemblies. Each partial assembly shall be composed of either all dummy or all fueled plates. A full assembly shall be replaced with no fewer than 13 plates in a pair of partial assemblies.

Fuel assemblies shall conform to nominal specifications presented in Table 5-1.

The reactor core shall be loaded so that all fuel assembly positions are occupied.

The fuel assemblies are contained in six aluminum boxes arranged in two parallel rows of three boxes each, separated by about 30 cm of graphite. The fuel boxes are surrounded by a 5 ft x 5 ft x 5 ft reactor grade graphite assembly.

The tops of the fuel boxes are covered during operations at power above 1 kW, by the use of the shield plugs and/or gasketed aluminum covers secured to the top of the fuel boxes. The devices function to prevent physical damage of the fuel, to minimize evaporation / leakage of water from the top of the fuel boxes, and to minimize entrapment of argon in the coolant water for radiological protection purposes.

5.5 Reactor Control and Safety Systems

Design features of the components of the reactor control and safety systems that are important to safety, as specified under Section 3.2 of these Technical Specifications, are given below.

5.5.1 Reactor Control System

Reactivity control of the UFTR is provided by four control blades, three safety blades and one regulating blade. The control blades are of the swing-arm type consisting of four aluminum vanes tipped with cadmium, protected by magnesium shrouds. They operate in a vertical arc within the spaces between the fuel boxes. Blade motion is limited to a removal time of at least 100 seconds and the insertion time under trip conditions is stipulated to be less than 1.5 sec. The reactor blade withdrawal interlock system prevents blade motion which will exceed the reactivity addition rate of 0.06% $\Delta k/k$ per sec, as specified in these Technical Specifications. The control blade drive system consists of a two-phase fractional horsepower motor that operates through a reduction gear train, and an electrically energized magnetic clutch that transmits a motor torque through the control blade shaft, allowing motion of the control blades. The blades are sustained in a raised position by means of this motor, acting through the electromagnetic clutch. Interruption of the magnetic current results in a decoupling of the motor drive from the blade drive shaft, causing the blades to fall back into the core. Position indicators, mechanically and electronically geared to the blade drives, transmit blade position information to the operator at the control console. Reactor shutdown can also be accomplished by voiding the moderator/coolant from the core. Two independent means of voiding the moderator/coolant from the core are provided:

(1) water dump via the primary coolant system dump valve opening under full trip conditions.

(2) water dump via the rupture disk breaking under pressure conditions above design value.

The integral worths of the individual safety blades vary from about 1.3 to 2.0 $\Delta k/k$ depending on position in the core and individual characteristics. The regulating blade worth is about 0.6% - 0.8% $\Delta k/k$. The blade worths, drive speeds, and drop-time values are sufficiently conservative to ensure compliance with the specified reactivity limitations. Additional reactivity and power related features are obtained from the control blade withdrawal inhibits. The regulating blade may be engaged by a servo-mechanism controlled by the linear channel for automatic reactor power control.

5.5.2 Reactor Safety System

(1) Power Level Channels

Two independent measuring channels are provided for power level limits; both are required for the reactor to be operable. Each channel covers reliably the range from about 1 to 150% of full power (of 100 kW). One channel (Safety 1) is part of the wide range drawer, and receives its main signal from a fission chamber. The Safety 2 channel uses an uncompensated ion chamber for neutron detection. Each channel drops all control blades and the moderator coolant from the core by actuating bistable trips in the safety system in a one-out-of-one trip logic. Visual indication of the power measured by each chamber, as well as annunciation of channel status is available to the operator in the control room.

(2) Wide Range Logarithmic Power Level and Period Channel

The logarithmic power channel covers the wide range from reactor startup to full power in 10 decades. It uses a fission chamber for this entire range and uses a B-10 proportional counter only in the startup (source) range. Signals from the fission chamber and the B-l0 counter are amplified by a preamplifier before going to the log channel. The preamplifier also processes test signals from the console controls and deenergizes the B-10 proportional counter at about 400 cps. Power level information is displayed on a meter and on a two-pen recorder. The channel provides the following blade withdrawal inhibits or blade trips: minimum source count inhibit of 2 cps, fast period inhibit of 10 sec, fast period trip of 3 sec, and inhibit limiting power escalation in the automatic mode to no faster than 30 sec period, and a trip at or above 1% power when secondary coolant flow is below the trip setting. Because this is a wide range channel, a separate startup channel is not used. These control or limiting actions prevent startup or operation of the reactor unless it is properly monitored or if operational restrictions are not met. Period is displayed on a meter and is effective for control over the entire range of operation.

(3) Startup (Neutron) Source(s)

A permanent, regenerable, antimony-beryllium source of up to 25 Ci and/or a removable plutonium-beryllium source of 1 Ci may be used for reactor startup to monitor the approach to criticality. The use of a neutron source ensures that behavior of the reactor is being monitored by the reactor instrumentation during subcritical control blade manipulations.

(4) Linear Neutron Channel and Automatic Flux Control System

The linear channel is required to be operable when the reactor is to be operated in the automatic mode. The linear channel uses a compensated ion chamber for neutron detection; its signal is transmitted by a multirange picoammeter. The picoammeter sends a signal to the linear channel of the two-pen recorder to display power level from source level to full power. It also sends a signal to the automatic flux controller which, in comparison with a signal from a percent of power setting control acts to establish and/or hold power level at a desired value. The rate of power increase is controlled by the action of a limiter in the linear channel/ automatic control system which maintains the reactor period at or slower than 30 sec. The automatic flux controller is not required to be operable for reactor operations where it is not needed and not to be used.

5.6 Cooling System

5.6.1 Primary Cooling System

The primary coolant is demineralized light water, which is normally circulated in a closed loop. The flow is from the 200-gal storage (dump) tank to the primary coolant pump; water is then pumped through the primary side of the heat exchanger and to the bottom of the fuel boxes, upward past the fuel plates to overflow pipes located about 6 in. above the fuel, and into a header for return to the storage tank. A purification loop is used to maintain primary water quality. The purification loop pump circulates about 1 gpm of primary water, drawn from the discharge side of the heat exchanger, through mixed-bed ion-exchange resins and a ceramic filter. The purification loop pump automatically shuts off when the primary coolant pump is operating, since flow through the purification system is maintained. Primary coolant may be dumped from the reactor fuel boxes by opening an electrically operated solenoid dump valve, which routes the water to the dump tank. A pressure surge of about 2 psi above normal in the system will also result in a water dump by breaking a graphite rupture disc in the dump line. This drains the water to the primary equipment pit floor actuating an alarm in the control room. The primary coolant system is instrumented as follows:

(1) thermocouples at each fuel box and the main inlet and outlet (eight total), alarming and recording in the control room; seven are required (core main inlet and core outlet on all six individual fuel boxes);

(2) a flow sensing device in main inlet line, alarming and displayed in the control room;

(3) a flow sensing device (no flow condition) in the outlet line, alarming in the control room;

(4) resistivity probes monitoring the inlet and outlet reactor coolant flow, alarming and displayed in the control room;

(5) an equipment pit water level monitor, alarming in the control room.

The reactor power is calibrated annually by the use of the coolant flow and temperature measuring channels.

5.6.2 Secondary Cooling System

Two secondary cooling systems are normally operable in the UFTR: a well water secondary cooling system and a city water secondary cooling system. Either system meets the requirements for secondary cooling. The well secondary cooling system is the main system used for removal of reactor generated heat to the environment. A deep well furnishes about 200 gpm of cooling water to the shell side of the heat exchanger, removing primary heat and rejecting it to the storm sewer. Weekly samples monitor the activity of this water. Flow indications in the control room are 140 gpm as a warning and 60 gpm to initiate a trip at or above 1 kW after an approximately 10-sec warning. The city water secondary cooling system can be used for backup cooling or for specific operations requiring reactor coolant temperatures hotter than those obtained with the well cooling system. Operability of this city water system is not a limiting condition for operation unless it is to be used for reactor operation at or above 1 kW. The secondary flow rate by the city water system is about 30 - 70 gpm, with a reactor trip set at 8 gpm (as measured by a flow switch) for power levels at or above 1 kW with approximately a 10 sec delay only upon first reaching 1 kW. A back flow preventer in the city water line ensures compliance with the requirements of the National Plumbing Code to prevent contamination of the potable water supply. The secondary coolant system inlet and outlet temperatures are monitored by tlermocouples, with recording and alarm functions in the control room.

5.7 Radiological Safety Design Features

5.7.1 Physical Features

The confinement structure consists of the reactor cell, with a free air volume of about 1600 m^3. This structure houses the reactor, reactor control room, the primary cooling system (including the dump tank, heat exchanger and purification loop), secondary coolant piping, and reactor vent system. Access to the reactor cell, which is the designated restricted and security area, is controlled by the specifications established by the Physical Security Plan of the UFTR.* Ventilation is through the independent air handler/ventilation and reactor vent system. The reactor vent system can be secured to

* Withheld from public disclosure pursuant to 10 CFR 2.790(d).

prevent uncontrolled discharge of radioactivity to the environment or releases in excess of permissible levels (per 10 CFR 20). Rough and absolute filters are used to eliminate or minimize radioactive air particulate contamination from the exhaust air. The electrically actuated damper in the core exhaust line is fail-safe and closes upon deenergization.

5.7.2 Monitoring System

Area and stacker radiation monitors are used for radioactivity monitoring, as delineated in Sections 3.5.1, 3.5.2, and 3.5.4 of these Technical Specifications. The cell air is monitored by an air particulate detector. Exhaust air drawn from the reactor cavity, reactor cell, or experiments is continuously monitored for gross concentration of radioactive gases and/or airborne radioactivity.

5.7.3 Evacuation Sequence

The emergency evacuation sequence is initiated either automatically by two area monitors alarming high in coincidence or manually by the console reactor operator. The sequence is that the reactor room air handler/ventilation system and the reactor vent system are shut down and the core vent damper is closed.

5.8 Fuel Storage

5.8.1 New Fuel

Unirradiated new fuel elements are stored in a vault-type room security area equipped with intrusion alarms in accordance with the Physical. Security Plan. Elements are stored in a steel, fireproof safe in which a cadmium plate separates each layer of bundles to ensure subcriticality under optimum conditions of moderation and reflection.

5.8.2 Irradiated Fuel

Irradiated fuel is stored upright and dry in storage pits within the reactor building in criticality-safe holes.

Table 5-1 Fuel Elements Nominal Specifications

Item	Specification
Overall size (bundle)	2.845 in. x 2.26 in. x 25.6 in.
Clad thickness	0.015 in.
Plate thickness	0.050 in.
Water channel width	0.111 in.
Number of plates	standard fuel element - 14 fueled plates partial element - no fewer than 13 plates in a pair of partial assemblies
Plate attachment	bolted with spacers
Fuel content per plate	12.5g U-235 nominal

6.0 ADMINISTRATIVE CONTROLS

6.1 Organization

6.1.1 Structure

The organization for the management and operation of the reactor facility shall include the structure indicated in Figure 61. Job titles are shown for illustration and may vary. Four levels of authority are provided.

Level 1 - individuals responsible for the reactor facility's licenses, charter, and site administration.

Level 2 - individual responsible for reactor facility management.

Level 3 - individual responsible for reactor operations, and supervision of day-to-day facility activities.

Level 4 - reactor operating staff (Senior Reactor Operator, Reactor Operator and trainees).

The Reactor Safety Review Subcommittee is appointed by, and shall report to, the Chairman of the Radiation Control Committee. The Chairman of the Radiation Control Committee reports to the Director of Environmental Health and Safety, who reports to the Vice-President for Finance and Administration. Radiation safety personnel shall report to Level 2 or higher.

6.1.2 Responsibility

Responsibility for the safe operation of the reactor facility shall be with the chain of command established in Figure 61. Individuals at various management levels, in addition to having responsibility for the policies and operation of the reactor facility, shall be responsible for safeguarding the public and facility personnel from undue radiation exposures and for adhering to all requirements of the operating license, charter, and technical specification. In all instances, responsibilities of one level may be assumed by designated alternates or by higher levels, conditional upon appropriate qualifications.

6.1.3 Staffing

The minimum staffing when the reactor is not secured shall be as follows:

(1) A certified reactor operator shall be in the control room.

(2) A second person shall be present at the facility complex able to carry out prescribed written instructions including instructions to initiate the first stages of the emergency plan, including evacuation and initial notification procedures. Unexpected absence for two hours is acceptable provided immediate action is taken to obtain a replacement.

(3) A designated Senior Reactor Operator (Class A Reactor Operator) shall be readily available on call. "Readily Available on Call" means an individual who:

  (a) has been specifically designated and the designation known to the operator on duty,
  (b) keeps the operator on duty informed of where he/she may be rapidly contacted and the phone number or other means of communication available, and
  (c) is capable of getting to the reactor facility within a reasonable time under normal conditions (e.g., 30 min or within a 15 mi radius).

A list of reactor facility personnel by name and telephone number shall be readily available in the Control Room for use by the operator. The list shall include:

(1) management personnel,
(2) radiation safety personnel, and
(3) other operations personnel.

Events requiring the presence of a Senior Reactor Operator are:

(1) all fuel or control-blade relocations within the reactor core region,

(2) relocation of any incore experiment with a reactivity worth greater than one dollar, and

(3) recovery from unplanned or unscheduled shutdowns.

6.1.4 Selection and Training of Personnel

The selection, training, and requalification of operations personnel shall meet or exceed the requirements of the American National Standard for Selection and Training of Personnel for Research Reactors, ANSI/ANS-15.4-1988, Section 4.

6.2 Review and Audit

A method for the independent review and audit of the safety aspects of reactor facility operations shall be established to advise management. The review and audit functions of the UFTR operations are conducted by the Reactor Safety Review Subcommittee (RSRS).

6.2.1 Composition and Qualifications

The RSRS shall be composed of a minimum of five members, including the Reactor Manager and Radiation Control Officer (both ex-officio voting members), the Chairman of the Nuclear and Radiological Engineering Department and two other members having expertise in reactor technology and/or radiological safety.

6.2.2 Charter and Rules

The review and audit functions shall be conducted in accordance with the following established charter:

Designation - The name of the Subcommittee is Reactor Safety Review Subcommittee (RSRS).

Accountability - The RSRS is a Subcommittee of and reports to the University Radiation Control Committee (URCC). The URCC provides radiological safety recommendations to the Director of Environmental Health and Safety.

Scope - The RSRS shall be responsible for the review of safety-related issues pertaining to the University of Florida Training Reactor (UFTR).

Purpose - The purpose of the RSRS is to ensure the safe operation of the UFTR through the discharge of the Subcommittee review and audit function.

Membership

(a) The RSRS shall consist of at least five members. Membership will include the Chairman of the Nuclear and Radiological Engineering Department, University Radiation Control Officer, Reactor Manager and two technical personnel familiar with the operation of reactors and with the design of the UFTR and radiological safety, at least one of whom is from outside the Department of Nuclear and Radiological Engineering. The two technical personnel will be recommended to the Chairman of the URCC by the Chairman of the Department of Nuclear and Radiological Engineering. Any member may designate a duly qualified representative from a standing URCC approved list to act in their absence.

(b) An Executive RSRS Committee will consist of the Reactor Manager, University Radiation Control Officer and Chairman of the RSRS.

(c) The Chairman of the RSRS will be appointed by the Chairman of the URCC. The Chairman of the RSRS is an ex-officio voting member of the URCC and will serve as liaison between the RSRS and the URCC.

(d) Members appointed to the RSRS shall be reviewed, and as appropriate, new appointments made by January 1 of each calendar year.

Meetings

(a) At least one meeting shall be held quarterly at intervals not to exceed 4 months. Meetings may be held more frequently as circumstances warrant, consistent with the effective monitoring of facility operations as determined by the RSRS Chairman.

(b) Review of draft minutes will be completed before subsequent meetings, at which time they will be submitted for approval. Responsibility to ensure that this is done falls upon the RSRS Chairman. The RSRS Chairman is charged with the responsibility to assure that the minutes are submitted for approval in a timely manner.

(c) A quorum shall consist of at least three members (and 50% or more of the RSRS membership) and at least three members must agree when voting, regardless of the number present.

6.2.3 Review Function

The following items shall be reviewed:

(a) determination that proposed changes in equipment, systems, tests, experiments, or procedures do not involve an unreviewed safety question;

(b) all new procedures and major revisions thereto having safety significance, proposed changes in reactor facility equipment or systems having safety significance;

(c) all new experiments or classes of experiments that affect reactivity or result in the release of radioactivity;

(d) proposed changes in technical specifications, license, or charter;

(e) violations of technical specifications, license, or charter;

(f) violations of internal procedures or instructions having safety significance;

(g) operating abnormalities having safety significance;

(h) reportable occurrences;

(i) audit reports and annual facility reports.

A written report or minutes of the findings and recommendations of the review group shall be submitted to RSRS members in a timely manner after the review has been completed and to the Chairman of the Radiation Control Committee whenever a finding is deemed to require review by Level 1.

6.2.4 Audit Function

The audit function shall include selective (but comprehensive) examination of operating records, logs, and other documents. Where necessary, discussions with cognizant personnel shall take place. In no case shall the individual immediately responsible for the area, audit in the area. The following items shall be audited:

(a) facility operations for conformance to the technical specifications and applicable license or charter conditions, at least once per calendar year (interval between audits not to exceed 15 months).

(b) the requalification and recertification program for the operating staff, at least once every other calendar year (interval between audits not to exceed 30 months).

(c) the results of action taken to correct those deficiencies that may occur in the reactor facility equipment, systems, structures, or methods of operations that affect reactor safety, at least once per calendar year (interval between audits not to exceed 15 months).

(d) the reactor facility emergency plan, and implementing procedures at least once every other calendar year (interval between audits not to exceed 30 months).

Deficiencies uncovered that affect reactor safety shall immediately be reported to the Radiation Control Committee and the Dean of the College of Engineering. A written report of the findings of the audit shall be submitted to the Dean of the College of Engineering and the review and audit group members within three (3) months after the audit has been completed.

6.3 Radiation Safety and ALARA (As Low As Reasonably Achievable)

The Radiation Control Committee and the Radiation Control Officer shall be responsible for the implementation of the Radiation Control Program for the UFTR. The primary purpose of the program is to assure radiological safety for all University personnel and the surrounding community.

The principal routine emission from the UFTR facility complex is Argon-41 discharged by the reactor vent system. There is little biological uptake of argon-41 and exposure limits are based upon external, total body irradiation.

The concentration of Argon-41 in the stack effluent is continuously monitored when the reactor is operating, and is normally less than 1×10^{-5} *μCi/ml* after several hours of full power operation. The annual release is related to the number of equivalent hours of 100 kW operation (kWth per year). Reactor operations are limited by prior agreement, and by these Technical Specifications, to limit argon-41 discharges to the maximum allowed concentration when averaged over a month and using the established atmospheric dilution factor of 200.

The offsite environmental radioactive surveillance program has proven that exposure to the general public from the reactor radioactive effluents consistently approaches the nondetectable level and certainly is always well below the 100 mrem/yr limit.

The ALARA program at the UFTR minimizes unnecessary production of radioactive effluents by selectivity of operations. The potential reduction of argon-41 releases is frequently reviewed, and was a major item of consideration during reviews to upgrade facility operations to 500 kWth. A reduction of the vent flow as well as the argon dissolving in the primary coolant has been proposed in the past, as well as the possibility of utilizing storage tanks.

Radioactive liquid effluents and personnel radioactive exposure are well within ALARA guidelines.

6.4 Procedures

The UFTR facility shall be operated and maintained in accordance with approved written procedures. All procedures and major revisions thereto shall be reviewed and approved by the Director of Nuclear Facilities and the RSRS before becoming effective.

The following types of written procedures shall be maintained:

(1) normal startup, operation and shutdown procedures for the reactor to include applicable checkoff lists and instructions;

(2) fuel loading, unloading, and movement within the reactor;

(3) procedures for handling irradiated and unirradiated fuel elements;

(4) routine maintenance of major components of systems that could have an effect on reactor safety;

(5) surveillance tests and calibrations required by the technical specifications or those that may have an effect on reactor safety;

(6) personnel radiation protection, consistent with applicable regulations;

(7) administrative controls for operations and maintenance and for the conduct of irradiations and experiments that could affect reactor safety or core reactivity;

(8) implementation of the Emergency Plan;

(9) procedures that delineate the operator action required in the event of specific malfunctions and emergencies;

(10) procedures for flooding conditions in the reactor facility, including guidance as to when the procedure is to be initiated and guidance on reactivity control.

Substantive changes to the above procedures shall be made effective only after documented review by the RSRS and approval by the facility director (Level 2) or designated alternates. Minor modifications to the original procedures which do not change their original intent may be made by the reactor manager (Level 3) or higher, but modifications must be approved by Level 2 or designated alternates within 14 days. Temporary deviations from the procedures may be made by a senior reactor operator, in order to deal with special or unusual circumstances or conditions. Such deviations shall be documented and reported to Level 2 or designated alternates.

6.5 Experiment Review and Approval

(1) Experiment review and approval shall be conducted as specified under Section 3.6, "Limitations on Experiments", of these Technical Specifications.

(2) Experiment review and approval shall ensure compliance with the requirements of the license, Technical Specifications, and applicable regulations and shall be documented.

(3) Substantive changes to previously approved experiments with safety significance shall be made only after review by the RSRS, approval in writing by Level 2 or designated alternates. Minor changes that do not significantly alter the experiment may be approved by Level 3 or higher.

(4) Approved experiments shall be carried out in accordance with established approved procedures.

6.6 Required Actions

6.6.1 Action to be Taken in Case of Safety Limit Violation

(1) The reactor shall be shut down, and reactor operations shall not be resumed until authorized by the Nuclear Regulatory Commission.

(2) The safety limit violation shall be promptly reported to Level 2 or designated alternates.

(3) The safety limit violation shall be reported to the Nuclear Regulatory Commission.

(4) A safety limit violation report shall be prepared. The report shall describe the following:

(a) applicable circumstances leading to the violation including, when known, the cause and contributing factors;

(b) effect of the violation upon reactor facility components, systems, or structures and on the health and safety of personnel and the public;

(c) corrective action to be taken to prevent recurrence.

The report shall be reviewed by the RSRS and any followup report shall be submitted to the Commission when authorization is sought to resume operation of the reactor.

6.6.2 Action To Be Taken in the Event of an Occurrence of the Type Identified in Section 6.7.2(2) and 6.7.2(3).

(1) Reactor conditions shall be returned to normal or the reactor shall be shut down. If it is necessary to shut down the reactor to correct the occurrence, operations shall not be resumed unless authorized by Level 2 or designated alternates.

(2) Occurrence shall be reported to Level 2 or designated alternates and to the Commission as required.

(3) Occurrence shall be reviewed by the review group (RSRS) at their next scheduled meeting.

6.7 Reports

In addition to the requirements of the applicable regulations, reports shall be made to the Nuclear Regulatory Commission as follows:

6.7.1 Operating Reports

Routine annual reports covering the activities of the reactor facility during the previous calendar year shall be submitted to the Commission within nine (9) months following the end of each prescribed year. The prescribed year ends August 31 for the UFTR. Each annual operating report shall include the following information:

(1) a narrative summary of reactor operating experience including the energy produced by the reactor and the hours the reactor was critical;

(2) the unscheduled shutdowns including, where applicable, corrective actions taken to preclude recurrence;

(3) tabulation of major preventive and corrective maintenance operations having safety significance;

(4) tabulation of major changes in the reactor facility and procedures, and a tabulation of new tests or experiments, that are significantly different from those performed previously and are not described in the Safety Analysis Report, including conclusions that no unreviewed safety questions were involved;

(5) A summary of the nature and amount of radioactive effluents released or discharged to the environs beyond the effective control of the facility operators as determined at or before the point of such release or discharge. (The summary shall include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25% of the concentration allowed, a statement to this effect is sufficient.);

(6) A summarized result of environmental surveys performed outside the facility;

(7) A summary of exposure received by facility personnel and visitors where such exposures are greater than 25% of that allowed.

The annual report shall be submitted with a cover letter to:

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

6.7.2 Special Reports

There shall be a report not later than the following working day by telephone and confirmed in writing by telegraph or similar conveyance to the Commission, to be followed by a written report that describes the circumstances of the event within 14 days of any of the following:

(1) Violation of safety limits (see Section 6.6.1);
(2) Release of radioactivity from the site above allowed limits (see Section 6.6.2);

(3) Any of the following: (see Section 6.6.2)

(a) Operation with actual safety-system settings for required systems less conservative than the limiting safety-system settings specified in the Technical Specifications;

(b) Operation in violation of limiting conditions for operation established in the Technical Specifications unless prompt remedial action is taken;

(c) A reactor safety system component malfunction that renders the reactor safety system incapable of performing its intended safety function, unless the malfunction or condition is discovered during maintenance test or periods of reactor shutdowns;*
*Note: Where components or systems are provided in addition to those required by the Technical Specifications, the failure of the extra components or systems is not considered reportable provided that the minimum number of components or systems specified or required perform their intended reactor safety function.

(d) An unanticipated or uncontrolled change in reactivity greater than one dollar (reactor trips resulting from a known cause are excluded);

(e) Abnormal and significant degradation in reactor fuel, or cladding, or both, coolant boundary, or confinement boundary (excluding minor leaks), where applicable, which could result in exceeding prescribed radiation exposure limits of personnel or environment or both;

(f) An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to reactor operations;

(g) A violation of the Technical Specifications or the facility license.

6.7.3 Other Special Reports

There shall be a written report sent to the Commission within 30 days of the following occurrences:

(1) permanent changes in the facility organization involving Level 1 (UF President, Dean of the College of Engineering, and Chairman of the Nuclear and Radiological Engineering Department), 2 or 3 personnel;

(2) significant changes in the transient or accident analyses as described in the UFTR Final Safety Analysis Report.

6.8 Records

Records of the following activities shall be maintained and retained for the periods specified below. The records may be in the form of logs, data sheets, computer storage media, or other suitable forms. The required information may be contained in single, or multiple records, or a combination thereof. Recorder charts showing operating parameters of the reactor (i.e., power level, temperature, etc,) for unscheduled shutdowns and significant unplanned transients including trips shall be maintained for a minimum period of 2 years.

6.8.1 Records To Be Retained for a Period of at Least Five Years

The following records are to be retained for a period of at least five (5) years:

(1) normal reactor facility operation (supporting documents such as checklists, log sheets, etc. shall be maintained for a period of at least 1 year);

(2) principal maintenance operations;

(3) reportable occurrences;

(4) surveillance activities required by the Technical Specifications;

(5) reactor facility radiation and contamination surveys where required by applicable regulations;

(6) experiments performed with the reactor;

(7) fuel inventories, receipts, and shipments;

(8) approved changes in operating procedures;

(9) records of meetings and audit reports of the RSRS.

6.8.2 Records To Be Retained for at Least One Training Cycle

Records of the most recent complete cycle of requalification and recertification training of certified operations personnel shall be maintained at all times the individual is employed.

6.8.3 Records To Be Retained for the Lifetime of the Reactor Facility

The following records are to be retained for the lifetime of the facility:

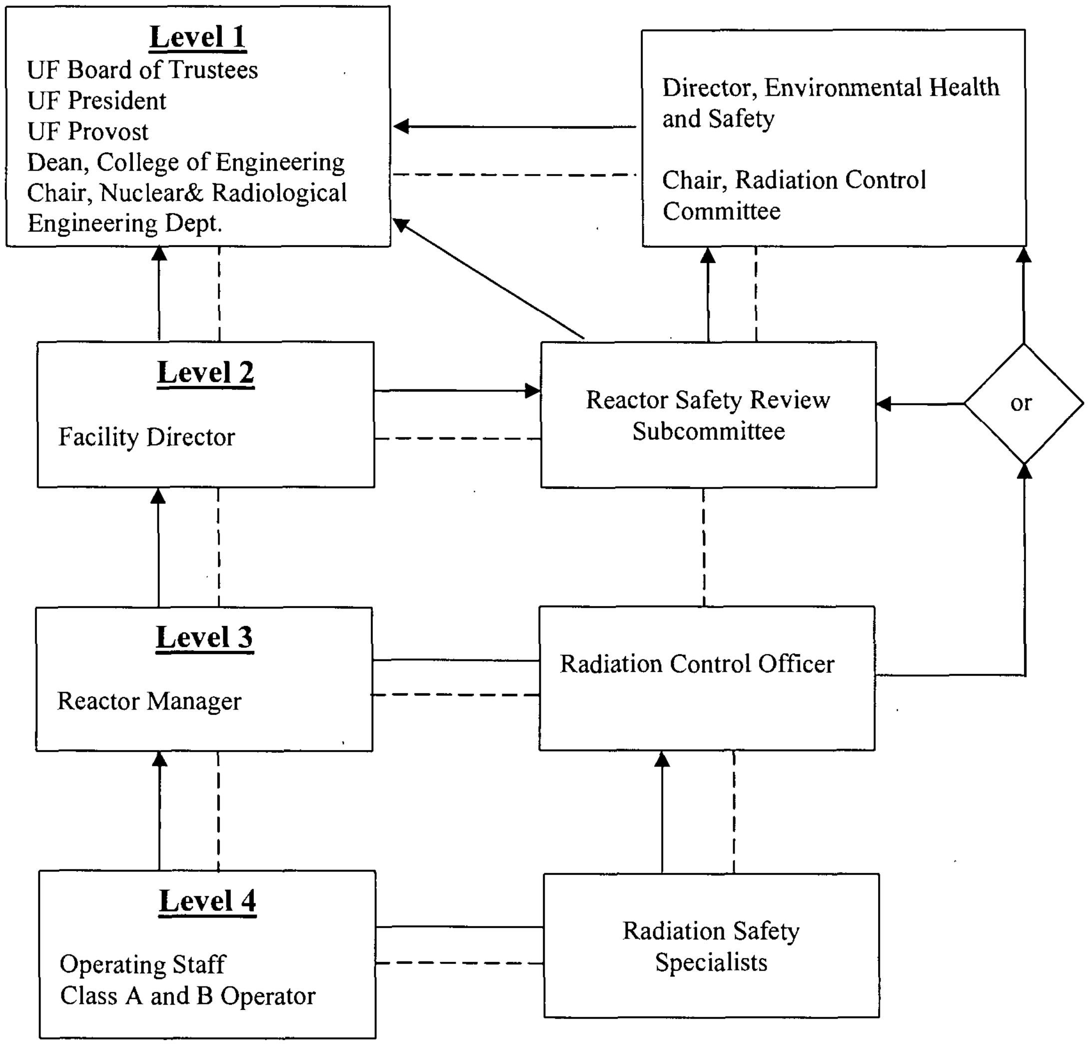
(1) gaseous and liquid radioactive effluents released to the environs;

(2) offsite environmental monitoring surveys required by the Technical Specifications;

(3) radiation exposure for all personnel monitored;

(4) updated drawings of the reactor facility.

Applicable annual reports, if they contain all of the required information, may be used as records in this section



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Reporting line: —— Communication line: ---- Reporting Responsibility: ⟶

Figure 6-1 UFTR Organizational Chart

SUBMITTAL REPORT
To Cover Analyses of
University of Florida Training Reactor (UFTR)
Conversion from HEU to LEU Fuel

Submitted by

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Second Version
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Summary
In this edition of the report, we combine the original submittal of the HEU-to-LEU conversion report with subsequent addendums submitted before receiving NRC approval. The mentioned addendums are listed below:

1. Addendum – 1: Request For Additional Information University of Florida Training Reactor Docket NO. 50- 83 (June 15, 2006)
2. Addendum – 2: University of Florida Training Reactor Supplemental Information (August 3, 2006)
3. Addendum – 3: University of Florida Training Reactor Supplemental Information (August 4, 2006)
4. Addendum – 4: University of Florida Training Reactor Supplemental Information: Effect of Handle On Margin to Onset of Nucleate Boiling (August 18, 2006)

*Note: Several changes also made to Technical Specifications in the Safety Analysis Report, not specified in this report.

This report contains the results of design and safety analyses performed by the University of Florida Nuclear and Radiological Engineering Department (NRE) and The Reduced Enrichment for Research and Test Reactors (RERTR) program at the Argonne National Laboratory (ANL) for conversion of the University of Florida Training Reactor (UFTR) from the use of highly-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel. This study investigates the performance and safety margins of the proposed LEU core under nominal and accident conditions. It identifies any necessary changes to the UFTR Final Safety Analysis Report and Technical Specifications (FSAR, Ref. 1).

1. General Description of the Facility

1.1 Introduction
This section provides an overview of the changes to the physical, nuclear and operational characteristics of the facility required by the HEU to LEU conversion of the UFTR fuel.

The HEU to LEU conversion only requires the use of a different fuel type and core configuration, and does not require any changes to the remainder of the facility.

The proposed LEU critical core contains 22 fuel bundles, one partial fuel bundle (with 10 fuel plates and 4 dummy plates), and one dummy bundle. Based on this core configuration, it is concluded: i) the shutdown margin meets the required limit; ii) the reactivity coefficients remain negative; iii) fuel integrity is maintained under all operating conditions; and iv) dose to public from the Maximum Hypothetical Accident (MHA) and Fuel Handling Accident (FHA) remains below the maximum permissible limit.

The HEU to LEU conversion requires changes to the Technical Specifications, procedures, and emergency plan as discussed in Sections 9, 12, and 14.

1.2 Summary and Conclusions of Principal Safety Considerations

The LEU core meets all the safety requirements as specified in FSAR.

1.3 Summary of Reactor Facility Changes

The LEU fuel bundle has the same overall design as the present HEU fuel bundle; except that it contains 14 fuel plates with U_3Si_2-Al fuel meat instead of 11 fuel plates of U-Al alloy fuel meat. The cladding of the HEU fuel is composed of 1100 aluminum alloy while the LEU fuel cladding is composed of 6061 aluminum alloy. This LEU silicide fuel has been approved by the Nuclear Regulatory Commission (NRC) for use in non-power reactors (Ref. 2).

1.4 Summary of Operating License, Technical Specifications, and Procedural Changes

In addition to the updated LEU fuel parameters, the safety limits presented in the Technical Specifications for power, flow rate and outlet temperature are changed (see Section 4.7 and 14) as well as the reactivity limitations on excess reactivity and experiment worth (see Section 13 and 14).

1.5 Comparison with Similar Facilities Already Converted

In 1991, the Iowa State University successfully converted their Argonaut reactor (UTR-10) facility using the same type of fuel plate. The main differences between the UTR-10 and UFTR are the power level and the core configuration. Following closure of the Iowa State reactor, fuel inspection revealed the presence of unexpected corrosion. This issue has been analyzed in an INL(Idaho National Lab)/ANL report (Ref. 3). To minimize the possibility of corrosion, the manufacturer (BWXT) of the LEU fuel for the UFTR will apply a surface treatment resulting in a protective boehmite layer on the surface of the cladding. It is concluded that corrosion should not occur in the UFTR core and, in fact, the corrosion of the fuel at the UTR-10 was not expected to limit core usage.

It is important to not that the Boehmite coating is extremely thin at 0.005 mm. It has essentially no effect on the finished dimensions of fuel plates. It therefore has no effect on hydraulic performance of the fuel. The plates will remain within the design tolerances. The heat conductivity of the boehmite coating is much less than that of the cladding (2.25 W/mC vs. 238.5 W/mC). Thus the effective clad conductivity is reduced from 238.5 to 97.1 W/mC for the LEU fuel design. For a typical operating point of 100 kW at 43 gpm and 30 C inlet temperature, the temperature drop across the clad is about 0.02 C with this effect included.

Although the INL/ANL report (Ref. 3) states the importance of maintaining primary coolant pH in a range of 5.4 to 6.0, the UFTR is less sensitive to pH range because it has much lower heat flux values than the higher power research reactors in Table 1 of the INL report and has water resistivity consistently above 1.0 megohm-cm. Therefore, it is only necessary to maintain the pH levels below 7.0. A phone conversation with Dr. Gerard Hoffman at ANL (Ref.3) on May 15, 2006 verified this.

The INL/ANL report (Ref. 3) also discusses the impact of not letting primary coolant stagnate in the core and not draining the core if water or air quality is poor. Such stagnation is not possible for the UFTR. The Primary Coolant (PC) storage tank and pump are located in a pit below floor level while the core is several feet above floor level. As a result, the only way to leave water in the core is to have the PC pump on; when it is off, the water automatically drains to the PC tank below floor level. Additionally, if water quality is poor (resistivity dropping near 1.0 megohm-cm via on line measurement plus a grab sample measurement weekly), the demineralizer resins are replaced to restore good water quality. So this condition is not allowed to continue.

2. Site Characteristics

The HEU to LEU conversion does not impact the site characteristics. More details about this topic can be found in Ref. 1.

3. Design of Structures, Systems, and Components

The HEU to LEU conversion does not require any changes to the design of structure, systems, and components. More details about this topic can be found in Ref. 1.

4. Reactor Description

4.1 Reactor Facility

The HEU to LEU conversion of the UFTR facility requires only changes in the core configuration and fuel type. All the following aspects of the facility remain unchanged:

- Control Blades
- Neutron Reflector
- Neutron Source and Holder
- In-Core Experimental Facilities
- Reactor Tank and Biological Shielding
- Core Support Structure
- Functional Design of the Reactivity Control System

The HEU and LEU cores contain different type of fuel meat, thickness of fuel meat and fuel plate, type of Al cladding, fuel enrichment, and fuel loading per plate. The current HEU core and the proposed LEU core also differ primarily in the number of fuel plates per fuel bundle with the number of full/partial fuel bundles, and number of dummy bundles. Note that the proposed LEU core configuration may differ when the actual fuel loading is performed.

Table 4-1 provides a comparison of the key design safety features of the HEU and LEU fuel bundles and a comparison of the key reactor and safety parameters that were calculated for each core. The results show that the UFTR reactor facility can be operated as safely with the new LEU fuel bundles as with the present HEU fuel bundles.

Table 4-1 Summary of Key Nominal Design Parameters of HEU and LEU Cores

	HEU	**LEU**
DESIGN DATA		
Fuel Type	U-Al alloy	U_3Si_2-Al
Fuel Meat Size		
Width (cm)	[redacted]	
Thickness (cm)		
Height (cm)		
Fuel Plate Size		
Width (cm)	[redacted]	
Thickness (cm)		
Height (cm)		
Cladding	1100 Al	6061 Al
Cladding Thickness (cm)	0.038	0.038
Fuel Enrichment (nominal)	93.0 %	19.75%
"Meat" Composition (wt% U)	14.05	62.98
Mass of ^{235}U per Plate (nominal)	[redacted]	
Number of Plates per Fuel Bundle	11	14
Number of Full Fuel Bundles (current/expected)	[redacted]	
Number of Partial Fuel Bundles	1 (5 fuel plates + 5 dummy plates)	1 (10 fuel plates + 4 dummy plates)
Number of Dummy Bundles	2	1
REACTOR PARAMETERS		
Fresh Core Excess Reactivity (% Δk/k)	1.09	0.925
Shutdown Margin (Δk/k)	3.11	3.17
Control blade worth,		
Regulating (% Δk/k)	0.87	0.65
Safety 1 (% Δk/k)	1.35	1.65
Safety 2 (% Δk/k)	1.63	1.81
Safety 3 (% Δk/k)	2.06	1.48
Maximum Reactivity Insertion Rate (% Δk/k/s)	0.042	0.045
Ave. Coolant Void Coefficient, (% Δk/k/%void)		
Fresh Core	-0.148	-0.153
Depleted Core		-0.146
Coolant Temp. Coefficient, (% Δk/k/°C)		
Fresh Core	-5.91E-03	-5.68E-03
Depleted Core		-5.26E-03
Fuel Temp. Coefficient, (% Δk/k/°C)		
Fresh Core	-2.91E-04	-1.65E-03
Depleted Core		-1.49E-03
Effective Delayed Neutron Fraction		
Fresh Core	0.0079	0.0077
Depleted Core		0.00756
Neutron Lifetime (μs)		
Fresh Core	187.4	177.5
Depleted Core		195.1

Table 4-1 Summary of Key Nominal Design Parameters of HEU and LEU Cores (continued)

THERMAL-HYDRAULIC PARAMETERS (100kW, 43 gpm[1], Tin=30 C)		
Max. Fuel Temperature [2] (°C)	66.5	64.5
Max. Clad Temperature [2] (°C)	66.5	64.4
Mixed Mean Coolant Outlet Temperature (°C)	40.8	40.5
Max. Coolant Channel Outlet Temp., (°C)	58.3	59.1
Minimum ONBR	1.98	2.09
Minimum DNBR	354	376

[1] This value corresponds to water channel spacing tolerance of 10 mils, while the rest of the calculations correspond to a water channel spacing tolerance of 20 mils, which relates to a flow rate of 48 gpm (see Section 4.7.3)
[2] At nominal operating conditions

4.2 Reactor Core

This chapter provides a detailed description of the components and structures in the reactor core. Comparisons between the HEU and LEU cores are presented when the conversion requires changes in some characteristics.

The UFTR is a heterogeneous, graphite/water moderated and water cooled reactor fueled with 19.75% enriched plate-type U_3Si_2-Al (LEU) fuel. In its current configuration, the core can contain up to 24 bundles (fuel or dummy) arranged in 2x2 arrays within six aluminum fuel boxes. It is possible to use bundles which contain a mix of dummy and fuel plates.

The reactor is controlled by means of four control blades (3 safety blades and 1 regulating blade) of swinging-arm type. The blades are mounted on the side of the core and swing downward through the core between the fuel boxes. Each control blade is encased in a magnesium shroud.

The fuel boxes and the magnesium shrouds are surrounded by a stack of graphite stringers, which act as both moderator and reflector. Figure 4-1 shows the UFTR core. More detailed figures of the UFTR core can be found in Ref. 1.

[1] This value corresponds to water channel spacing tolerance of 10 mils, while the rest of the calculations correspond to a water channel spacing tolerance of 20 mils, which relates to a flow rate of 48 gpm (see Section 4.7.3)[2]At nominal operating conditions
[2] At nominal operating conditions

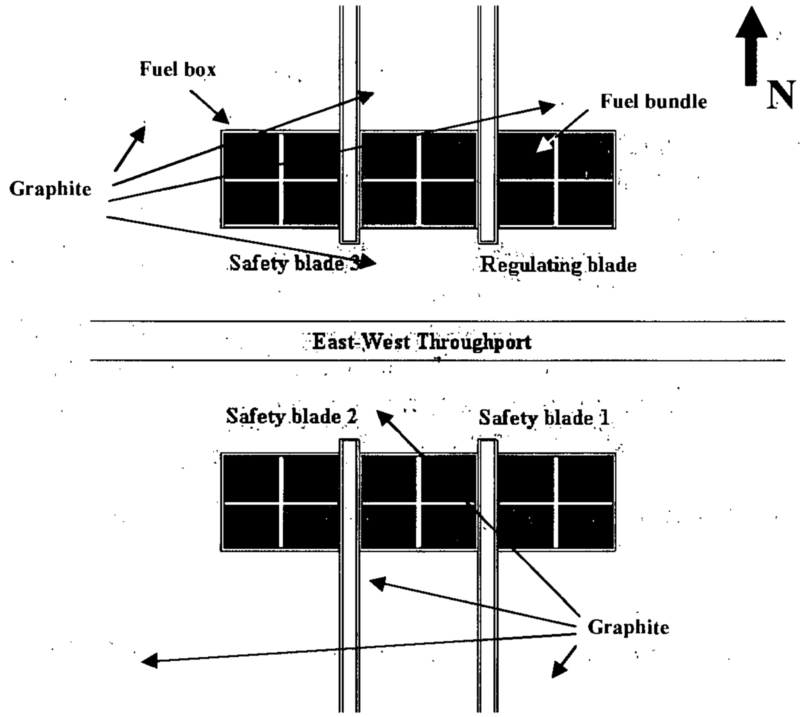


Figure 4-1 Schematic Horizontal Cut of the UFTR Core

Heat removal is achieved during reactor operation by providing forced circulation into the core, i.e., by pumping water upward through fuel boxes in a closed loop and consequently between the plates and fuel bundles contained in each box.

4.2.1 Fuel Elements

The HEU and LEU fuel elements have similar overall designs, i.e., they are both plate-type elements composed of a "sandwich" of fuel "meat" and aluminum cladding. The plates are then assembled in bundles which, in turn, are inserted into the fuel boxes. More details about the specifications of the UFTR LEU fuel can be found in Ref. 4.

Technical Specification 5.3 currently allows [redacted] kg of contained Uranium-235. Current actual possession value for contained U-235 is below [redacted] kg in the form of both HEU ([redacted] kg U-235) and LEU [redacted] kg U-235 some of which is unirradiated). For the expected 26 full 14-plate bundles with [redacted] g of U-235 per fuel plate, the additional U-235 could be [redacted] kg, allowing for plate uncertainties. Therefore, assume [redacted] kg bringing the total new possession limit to [redacted] kg plus some extra for fission chambers, flux foils, etc.

The following proposed possession limits apply for contained U-235 when the new LEU fuel is received.

Possession Limits on U-235 Upon Receipt of LEU Fuel

Kilograms	% Enrichment	Form
[redacted]	93%	Materials Test Reactor (MTR)-Type Fuel
	< 20%	MTR-Type Fuel
	Any %	Fission Chambers, Flux Foils and Other Forms Used in Connection with Operation of the Reactor

After all HEU irradiated fuel is removed from consideration, the first item in facility possession limit is deleted.

Fuel Plate Description

The HEU fuel meat consists of uranium-aluminum alloy with 93 wt% enriched uranium while the LEU fuel meat consists of U_3Si_2-aluminum dispersion fuel with 19.75 wt% enriched uranium. Table 4-2 compares various characteristic of the HEU and LEU fuel elements.

Table 4-2 Characteristics of the HEU and LEU Fuel Elements

	HEU [1]	**LEU**
Fuel plate, Width (cm) Thickness (cm) Height (cm)	[redacted]	
Fuel meat, Width (cm) Thickness (cm) Height (cm)		
Cladding, Along Width (cm) Along Thickness (cm)	 0.635 0.038	 0.635 [2] 0.038
Coolant Channel Thickness (cm)	0.348	0.282
Volume ratios, Fuel-to-Coolant	 0.08	 0.05

[1] UFTR drawing #021-80-107 which corresponds to the 1957 design

[2] Drawing #441597 from INEEL shows tolerances between [redacted]cm and [redacted]cm for the width of the fuel meat (x-axis). The size was chosen to be the same as HEU for convenience until an "as-built" drawing is available.

The major difference in the fuel plates is that the LEU fuel meat is one half the thickness of the HEU fuel meat. Figures 4-2 a) and b), respectively, show the HEU and LEU plates, and provide the dimensions of the different segments of each plate type.

[redacted]

Figure 4-2 HEU and LEU Fuel Plate Dimensions

The axes displayed in Figure 4-2 represent the orientation of the elements in the model. The x-axis, y-axis, and z-axis are set along the east-west, north-south and bottom-top axes of the core, respectively.

The HEU and LEU dummy plates are identical to their respective fuel plates with the obvious exception of the “meat”.

Fuel Bundle Description

Each HEU fuel bundle is composed of 11 fuel plates. A water gap of 0.348 cm is provided between the fuel plates for coolant flow as illustrated in Figure 4-3.

[redacted]

Figure 4-3 HEU Fuel Bundle XY Cut

The LEU fuel bundles are composed of 14 fuel plates with a water gap of 0.282 cm. Figure 4-4 depicts the layout of the fuel plates and water gaps in a LEU fuel bundle.

[redacted]

Figure 4-4 LEU Fuel Bundle XY Cut

Table 4-3 compares the bundle characteristics for HEU and LEU cores.

Table 4-3 Fuel Bundle Characteristics of the HEU and LEU Cores

	HEU	**LEU**
Number of Plates	11	14
Dimensions, x-axis (cm) y-axis (cm) z-axis (cm)	[redacted]	

[1] The HEU fuel bundle as defined in the FSAR includes half a fuel element spacing of water at each side of the bundle (see Figures 4-3 and 4-4). The actual bundle size, without this extra water, is [redacted] cm. The LEU fuel bundle was defined using the same approach.

Fuel Box Description

The existing fuel boxes are planned to be used for the LEU fuel. Therefore, the dimensions of the fuel boxes for the HEU and LEU cores are identical. Table 4-4 provides dimensions of a fuel box.

Table 4-4 Fuel Box Dimensions

Inner Fuel Box, along x-axis (cm) along y-axis (cm) along z-axis (cm)	 15.24 12.7 121.9
Fuel Box Wall Thickness (cm)	0.318
Fuel Box Spacing, along x-axis (cm) along y-axis (cm)	 2.54 30.48
½ Water Gap (between bundles), along x-axis (cm) along y-axis (cm)	 0.394 0.283

These dimensions are based on the current fuel size. The fuel region is vertically centered in the fuel box. Based on UFTR drawing #001-80-100, the water level is assumed to be at 5.08 cm below the top of the fuel box, i.e., at half the outlet pipe. This is confirmed by measurement of the water column height in the reactor building (measured at an average of 45.5" (115.57cm)). Figures 4-5 and 4-6 show the fuel box dimensions and the arrangement (as modeled) of the bundles inside the fuel boxes.

[redacted]

Figure 4-6 XY Cut of One Quarter of a Fuel Box

4.2.2 Control Blades

The UFTR reactivity control system consists of three safety blades (labeled S1 to S3) and one regulating blade (labeled SR) rotating in and out of the core region along a vertical arc within the space provided between the fuel boxes. A cadmium insert is located at the tip of each blade. These blades are protected by magnesium shrouds. The control blades and shrouds are not expected to be replaced, and therefore are identical for the HEU and LEU core. Figure 4-7 illustrates the location of the magnesium shrouds within the UFTR core.

[redacted]

Figure 4-7 Location of Control Blade Shrouds

The dimensions of the shroud are based on UFTR drawing #89-31-118. The blades have a fully-inserted nominal position of 2.5 degrees above the XY center plane and are moved out of the core by rotating them 45 degrees. The top of the shroud is located 10 cm above the top of the fuel box. Figure 4-8 shows the fully inserted and fully withdrawn location of the control blade with respect to one of the shrouds and the centerline of the core.

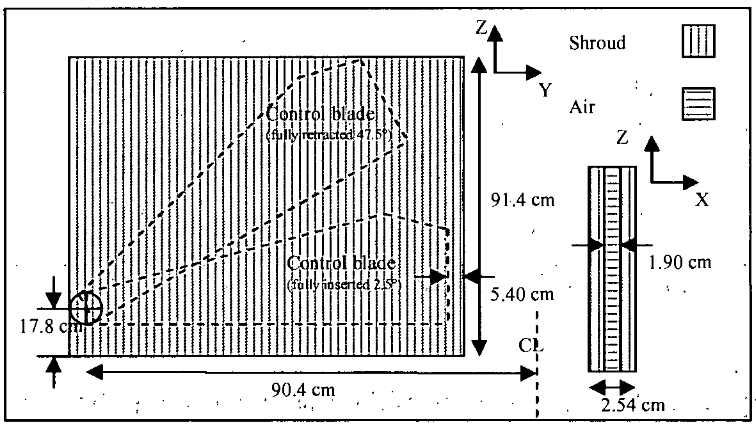


Figure 4-8 XZ and YZ Cut of a Magnesium Shroud

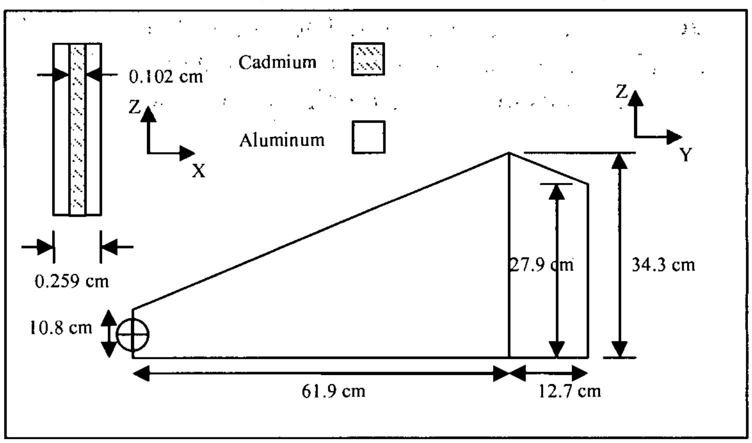


Figure 4-9 Control Blade Dimensions

Figure 4-9 shows the dimensions and location of the blades as given in the UFTR drawings #021-80-100, #021-80-102 and #021-80-113.

Figure 4-10 shows the dimensions of the cadmium inserts for both the safety blades (all nominally identical) and the regulating control blade. These dimensions are obtained from drawing #89-31-121 and from x-ray radiographic images of the blades.

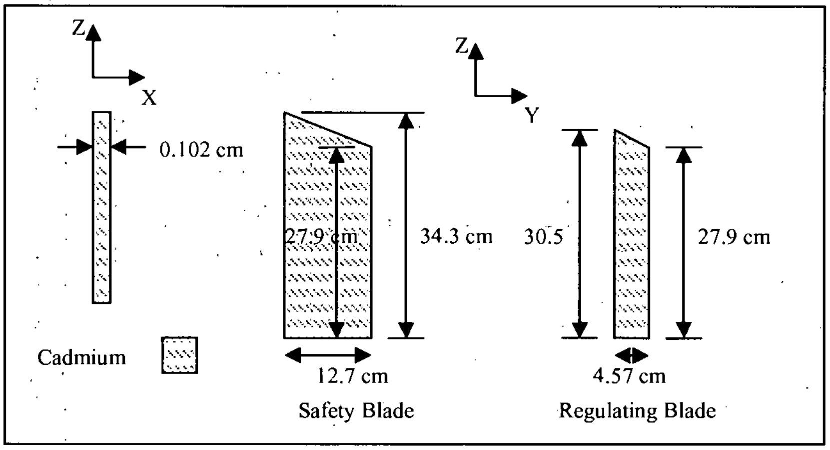


Figure 4-10 Cadmium Absorber Insert Dimensions

4.2.3 Neutron Reflector

Since there is no plan to change the reflector during the HEU to LEU conversion, the discussions presented here apply to both cores. The UFTR reactor uses nuclear-grade graphite (as well as water) as reflector. Figure 4-11 shows the different reflector regions as modeled.

[redacted]

Figure 4-11 Reflector Regions Modeled in MCNP5

4.2.4 Neutron Source and Holder

The proposed HEU to LEU conversion of the UFTR core does not require any changes in the existing neutron source location. More details about the characteristics of the current design can be found in Ref. 1.

4.2.5 In-Core Experimental Facilities

The proposed HEU to LEU conversion of the UFTR core does not require any changes to the in-core experimental facilities. More details about the experimental characteristics of the current design can be found in Ref. 1.

4.2.6 Reactor Materials

The UFTR conversion to LEU requires changing the fuel and cladding compositions. The LEU silicide fuel has been approved by NRC for use in non-power reactors. More detailed information can be found in Ref. 2. Table 4-5 compares the material compositions of the HEU and LEU fuel plates.

Table 4-5 Composition of the HEU and LEU Fuel

	HEU	LEU
Fuel "Meat"		
Composition	U-Al alloy	U_3Si_2-Al
Enrichment	93%	19.75%
Mass of ^{235}U per Fuel Plate	[redacted] [1]	[redacted]
Weight Fraction of Uranium (%)	14.05	62.98
Cladding		
Composition	1100 Al	6061 Al

[1] Average value (see Appendix A.1 for actual fuel loading)

4.3 Reactor Tank and Biological Shielding

The proposed HEU to LEU conversion of the UFTR core does not require any changes in the reactor tank or biological shielding. More details about this topic can be found in Ref. 1.

4.4 Core Support Structure

The proposed HEU to LEU conversion of the UFTR core does not require any changes in the core support structure. More details about this topic can be found in Ref. 1.

4.5 Dynamic Design

4.5.1 Calculation Model

In order to design the LEU core and determine the necessary operational and safety related parameters, a detailed calculational model for the HEU core was developed and benchmarked against experimental data. A similar model was then developed for the LEU core. These calculations utilized the MCNP5 (Ref. 5) Monte Carlo code with the ENDF/B-VI continuous energy cross section library (when these cross-sections are available, otherwise the latest cross-section library is used), and the SAS2 sequence of the SCALE5 package (Ref. 6) for fuel depletion calculations.

This section provides information on the material composition (fresh and depleted) for both cores, discusses the MCNP5 model developed for these analyses and the benchmark calculations for the HEU core, and determines a reference critical LEU core.

<u>Material Composition</u>

The HEU reactor core was modeled at two different burnups; beginning-of-life (BOL, fresh fuel) and current (depleted fuel at about 21.2 MWD for the oldest bundles). For the beginning-of-life core, the isotopic compositions and densities are presented in Table A.1-1, Appendix A.1.

To perform the required depletion calculations for the HEU core, the beginning-of-life peak-to-average ratios presented in Table A.1-2 (Appendix A.1) and the power histories presented in Appendix A.2 are used. Further discussions on determination of peak-to-average power ratios are presented in Appendix A.1.

Similar to the HEU core, two burnup states were also considered for the LEU core: BOL (fresh fuel); expected end-of-life (depleted fuel at about ~86.67 MWD; this is based on operation load of 4hr/day, 5day/week, 20 years at 100 kW). This power history results in a burnup that is about four times larger than the current HEU core.

The LEU U_3Si_2 fuel composition at BOL was obtained by averaging 6 sets of concentrations obtained from the manufacturer BWXT (Ref. 7). The fuel matrix aluminum alloy and aluminum cladding compositions were obtained from the same package. Further, it is important to note that in case the impurity concentration is not exact, rather bounded, we have used the maximum value. The detailed isotopic compositions and densities for the LEU core are presented in Tables A.1-3a, A.1-3b, and A.1-3c of Appendix A.1. The formulation for estimating fuel porosity was obtained from an IAEA document (Ref. 8).

For the fuel meat, the calculated density is 5.55 g/cm^3 with a ^{235}U loading of [redacted] fuel enrichment of 19.75 wt%. This means that the nominal mass of ^{235}U is [redacted] g per full fuel bundle. Table A.1-6 presents the peak-to-average power ratios at BOL used for the LEU depletion calculations.

As mentioned earlier, the depletion calculations required to model the current core were performed using the SAS2 sequence of the SCALE5 package which tracks a large number of fission products. However, it is only necessary to obtain the concentrations of the most important fission products, i.e., those that have the most effect on the reactivity of the core. The fission products were selected based on their poisoning ratio, i.e., the ratio of neutrons absorbed by the fission product to the neutrons absorbed by fuel. Consequently, in addition to the various uranium and plutonium isotopes, the highly neutron-absorbing fission products were considered as well as some long-lived isotopes. Table 4-6 presents the selected isotopes.

Table 4-6 Selected Isotopes Considered for Criticality Calculation [1]

Element	**Isotope**
Uranium	234, 235, 236, 238
Plutonium	239, 240, 241
Iodine	129, 131
Xenon	131, 133, 135
Samarium	149, 151
Promethium	147
Technetium	99
Neodymium	143, 145
Rhodium	103

[1] Due to the low burnup of the fuel and the spectrum characteristics of the UFTR, some of these isotopes may not be present in all the bundles.

Other materials used in the HEU and LEU cores are aluminum for cladding and other structures, graphite for moderator and reflector, cadmium tips for the control blades and magnesium for the control blade shrouds. Table 4-7 presents the characteristics of these various materials.

Table 4-7 Other Materials Characteristics for HEU and LEU Cores

	Composition		Density	
Material	HEU	LEU	HEU	LEU
Aluminum - cladding	Al + 10ppm of natural boron	See Table 4-8	2.70 g/cc	2.70 g/cc
Aluminum - other structures	Al + 10ppm of natural boron	Al + 10ppm of natural boron	2.70 g/cc	2.70 g/cc
Graphite -nuclear-grade	C + 5ppm of natural boron	C + 5ppm of natural boron	1.60 g/cc	1.60 g/cc
Cadmium (abundance in %) - natural cadmium	106Cd (1.25) 108Cd (0.89) 110Cd (12.49) 111Cd (12.80) 112Cd (24.13) 113Cd (12.22) 114Cd (28.73) 116Cd (7.49)	106Cd (1.25) 108Cd (0.89) 110Cd (12.49) 111Cd (12.80) 112Cd (24.13) 113Cd (12.22) 114Cd (28.73) 116Cd (7.49)	8.75 g/cc	8.75 g/cc
Magnesium	Mg	Mg	1.74 g/cc	1.74 g/cc

The 10ppm of natural boron-equivalent in the HEU aluminum cladding and structure material correspond to the best estimate of the impact of the impurities. The 5 ppm of natural boron-equivalent in the graphite corresponds to our best estimate of the impurities based on INL chemical analysis (Ref. 9) of several graphite samples.

Table 4-8 Composition of LEU Fuel Cladding

Isotope	Weight Fraction (%)
Al	97.599
Si	0.500
Fe	0.354
Cu	0.294
Mn	0.070
Mg	0.924
Cr	0.135
Zn	0.089
V	0.010
Zr	0.003
B	0.001
Co	0.001
Ga	0.005
Cd	0.001
Li	0.001

Geometric Model
Detailed MCNP5 models were developed for the HEU and LEU cores. Both models represent the reactor core, the moderator and reflector regions as well as part of the thermal column. MCNP5 capabilities allowed modeling of the geometry described in Section 4.2. The major differences between the geometric models of the HEU and LEU cores are the size and number of fuel plates, and the number of fuel bundles, dummy bundles and dummy plates.

Figures 4.12(a) and 4.12(b) show the axial and radial projections of the UFTR model.

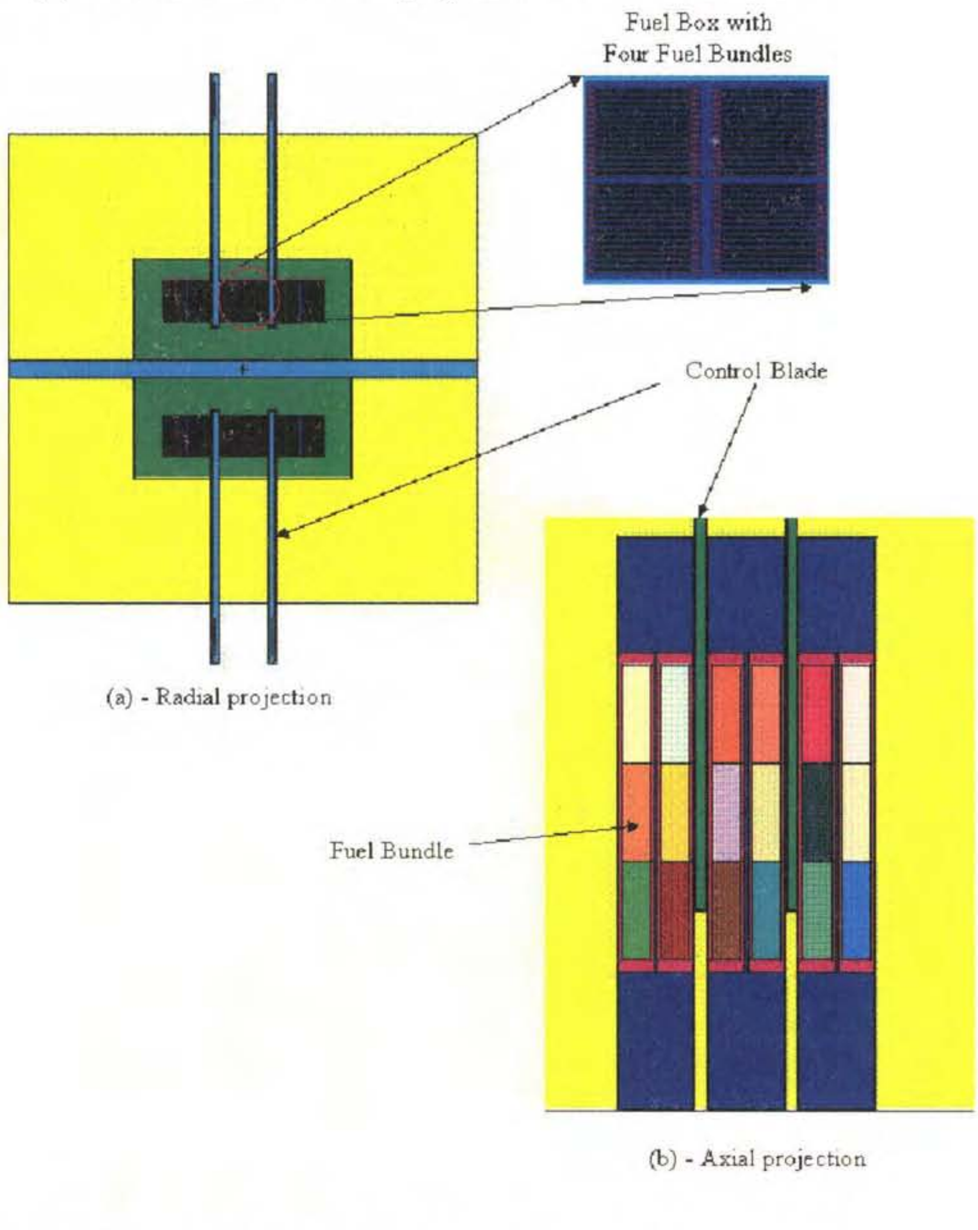
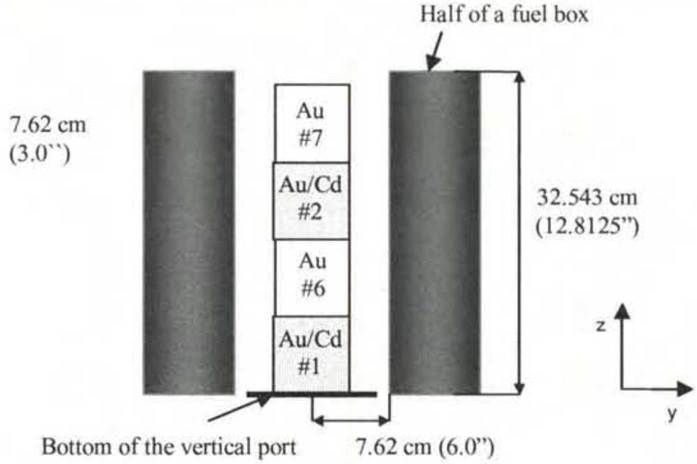


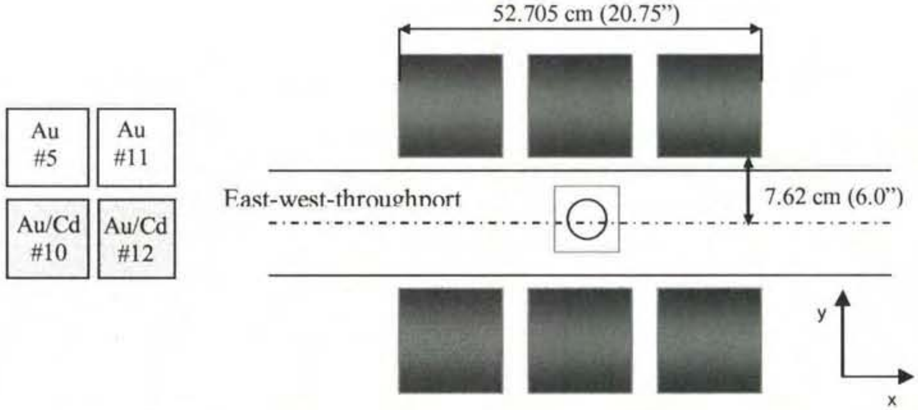
Figure 4-12 Schematic of the UFTR MCNP5 Model

Benchmarking of the HEU Core Criticality Model
In order to benchmark the HEU core model, experiments were performed by placing uncovered and cadmium-covered gold foils at the center vertical port and rabbit system of the UFTR.

Figure 4-13 shows the experimental setup for determination of reaction rates in the center vertical port (CVP) and the rabbit system which is located in the east-west throughport.



a) Schematic of the foil positions within the central vertical port



b) Top view of the rabbit system

Figure 4-13 Schematic of the Foil Positions within the CVP and Rabbit System

Gold and Cd-covered gold Au foils were placed alternately along the axis of the center vertical port. The foils used in the rabbit system were positioned at the center of the core, which is right below the bottom of the center vertical port.

Table 4-9 presents measured and calculated reaction rates in gold and Cd-covered gold foils located at different axial positions within the center vertical port of UFTR and the center of core within the rabbit system.

Table 4-9 Comparison of Measured and Calculated Foil Reaction Rates in the CVP and Rabbit System

Foil ID	Foil Type	Foil Position	Measured Reaction Rate	Calculated Reaction Rate (relative error)[3]	Ratio Measured to Calculated
1	Cd-covered Au	Vertical port	6.77E+09	6.78E+09 (7.47%)	1.00
6	Au	Vertical port	2.39E+10	2.43E+10 (3.79%)	0.98
2	Cd-covered Au	Vertical port	6.91E+09	5.80E+09 (8.25%)	1.19
7	Au	Vertical port	2.23E+10	1.82E+10 (4.70%)	1.23
10,12 [1]	Cd-covered Au	Rabbit system	5.96E+09 [2]	6.05E+09 (6.10%)	0.99
5,11 [1]	Au	Rabbit system	2.17E+10 [2]	2.29E+10 (3.22%)	1.06

[1]Measured twice at the same position with two foils.
[2]Averaged value for the two measurements at the same position
[3]1-σ statistical uncertainty

For the first two axial segments from the bottom of the central vertical port, the measured and calculated relative differences are within 1%, and for the next two axial positions the differences are within ~20%.

For the Cd-covered Au foil positioned at the center of the rabbit system, the measured and calculated reaction rates are within 1%. For the Au foil at the same position, the measured and calculated reaction rates are within 6%.

4.5.2 Critical Core Configuration

The current HEU core is composed of 21 fuel bundles, 1 partial fuel bundle and 2 dummy bundles. As mentioned in Section 4.2.1, each full fuel bundle contains 11 fuel plates, while the partial (or half fuel bundle) contains 5 fuel plates and only five dummy plates. The remaining two dummy bundles contain 11 dummy plates each. Figure 4.14 shows the pattern of the fuel and dummy bundles for the HEU core.

[redacted]

The two dummy bundles are located in the north-east and south-east corners, and the partial fuel bundle is located in the south-west corner.

To develop a reference LEU core, criticality calculations were performed for different core configurations. Based on these results, the reference critical LEU core is composed of 22 fuel bundles with 14 fuel plates each, 1 partial fuel bundle of 10 fuel plates and 3 dummy plates, and one dummy bundle with 14 dummy plates. Figure 4-15 shows the reference LEU core fuel pattern.

[redacted]

The partial fuel bundle and the dummy bundle are located at the south-east and north-east corners, respectively.

Due to various uncertainties, the as-built core can differ slightly from the proposed and analyzed core. Therefore, we have analyzed the impact of the number of plates in the partial fuel bundle 3-2 (Figure 4-15) on the bundle peak-to-average power ratio, the power per plate in the critical bundle, and the width profile in the critical plate. Based on these results (see Appendix A.7 for details), the impact on the thermal-hydraulics analysis for the LEU core is negligible because the changes in the power distributions are very small for a partial bundle with 6 and 13 fuel plates.

We expect that changes in fuel element location planned from the LEU reference core during the life of the core may occur toward the end of life, e.g., 20 years from now; so, it is not an issue for achieving criticality.

The positions of the control blades for the HEU critical core were measured and used to demonstrate that the MCNP5 model achieves a critical core. For the LEU core, the positions for the control blades were determined by performing criticality calculations. Table 4-10 compares the positions of the control blades for the HEU and LEU critical cores.

Table 4-10 Control blade positions for the HEU and LEU Cores

Control Blade	**Position (degree)**	
	HEU (measured)	**LEU (calculated)**
Safety 1 (SE)	38.5	26.3
Safety 2 (SW)	38.5	26.3
Safety 3 (NW)	38.5	26.3
Regulating (NE)	18.7	16.9

Since the LEU core contains fresh fuel, the safety blades are inserted farther into the core. The critical positions of the control blades for the LEU core are similar to the positions for the HEU core at BOL. The impact of impurities on the HEU core is discussed in Appendix A.3, and variations in the k_{eff} of the LEU for other core configurations are discussed in Appendix A.4. Table 4-11 compares the power distribution in each fuel bundle of the current HEU and reference LEU cores.

Table 4-11 Power Generated in Fuel for Depleted HEU and Reference LEU Cores

Bundle Number	Power (kW)		Relative Difference (%)
	HEU	LEU	
1-1	1.96	3.86	97.2
1-2	4.17	4.33	3.8
1-3	4.05	4.25	5.0
1-4	4.65	4.82	3.6
2-1	4.46	4.60	3.2
2-2	4.34	4.56	5.0
2-3	5.01	5.13	2.3
2-4	4.88	5.09	4.2
3-1	3.72	4.03	8.3
3-2	0.0 (dummy)	2.72	n/a
3-3	4.06	4.45	9.8
3-4	3.58	3.85	7.7
4-1	4.62	4.01	-13.3
4-2	5.21	4.49	-13.8
4-3	4.20	3.58	-14.6
4-4	4.75	3.96	-16.7
5-1	5.36	4.68	-12.8
5-2	5.01	4.45	-11.1
5-3	4.84	4.10	-15.4
5-4	4.52	3.91	-13.4
6-1	4.11	3.71	-9.7
6-2	3.64	3.36	-7.6
6-3	3.76	3.30	-12.2
Total[1]	**94.89**	**95.25**	**0.376**

[1] For the HEU and LEU cores, 5.11 kW and 4.75 kW, respectively, is deposited in the coolant, graphite moderator and core structures materials.

In the reference LEU core, bundle 1-1 (see Fig. 4-15 for its location) is a full fuel bundle instead of a half fuel bundle in the HEU core (see Fig. 4-14 for its location), the bundle 3-2 is a partly loaded fuel bundle with 10 fuel plates and 3 Al dummy plates, instead of a dummy bundle in the HEU core, and the bundle 6-4 is a dummy bundle in both HEU and LEU cores. The total power generated in the fuel of the LEU core is ~0.38% higher than in the fuel of HEU core. Further, there is a shift of power from north to south. Both cores have a total power of 100 kW. This is expected because the LEU core contains more fuel plates in the south part of core as compared to the HEU core.

Finally, the energy group flux profiles for the HEU and LEU cores are compared in Section A.5.

4.5.3 Excess Reactivity and Control Blade Worth

Excess Reactivity for HEU and LEU Cores

The MCNP5 code was used to calculate the excess reactivity for both fresh and depleted fuel for both HEU and LEU cores. The excess reactivity was evaluated by rotating the control blades to their fully withdrawn positions (47.5 degrees from horizontal) and calculating the corresponding k_{eff}.

Table 4-12 compares the calculated excess reactivity for the fresh and depleted HEU and LEU cores.

Table 4-12 Calculated Excess Reactivity for the HEU and LEU Cores (Fresh and Depleted)

Status	HEU ($\Delta k/k$ %)	Reference LEU ($\Delta k/k$ %)	Relative Difference (%)
Fresh	1.09	0.93	-15.1
Depleted	0.47	-0.42 [1]	n/a

[1] The power history used for the depletion calculation is different from the HEU power history. The LEU power history was selected in order to investigate the lifetime of the new core. (see Appendices A.2 and A.4). This implies that it will be necessary to insert additional fuel bundles in the available location before the end of this selected power history.

The calculated excess reactivity of the current core is 0.47% (+/- 0.03%). This value is consistent with the last measured excess reactivity of 0.38% performed in February 2005, but indicates a small bias of about 0.1% $\Delta k/k$, which can be partially attributed to experimental uncertainties.

Integral Control Blade Worth for HEU and LEU Cores

To evaluate the worth of each control blade, $\Delta k/k$ was calculated between the case where all the blades are fully withdrawn and the case where a given blade is fully inserted

Table 4-13 compares the worth of control blades as measured and calculated for the HEU depleted core, and for the fresh and depleted LEU core.

Table 4-13 Comparison of Control Blades Worth for the HEU and LEU Cores

Control Blade	HEU-depleted (calculated)	HEU-depleted (measured)	LEU-fresh (calculated)	LEU-depleted (calculated)
Regulating	0.87%	0.82%	0.63%	0.66%
Safety 1	1.35%	1.21%	1.62%	1.65%
Safety 2	1.63%	1.36%	1.81% [1]	1.76%
Safety 3	2.03% [1]	1.88%	1.42%	1.46%

[1] The total integral reactivity worth of this blade was evaluated by positioning all the blades at their critical position and then rotating the blade of interest (see following section).

For the HEU core, the calculated and experimental data differ in a range of 6.1% to 19.9%. These differences can be attributed to experimental uncertainty and inconsistency between the experimental procedure to measure the blade worth and the modeling procedure.

Further, the two control blades on the south part of the reference LEU core (safety 1 and 2) have higher worths as compared to the HEU core, while the two control blades on the north part of the LEU core (safety 3 and regulating) have lower worth than in the HEU core. This finding is expected because of the observed power shift presented in Table 4-10. This power shift is expected since more fuel is added to the south part of the core.

Maximum Reactivity Insertion Rate for HEU Core

In addition to calculations of the total reactivity worth for the UFTR control blades, an analysis of the integral worth as a function of position was performed for the most reactive blade. In the prior calculations, Safety Blade 3 was determined to be the most reactive blade. An MCNP model of the UFTR fueled with 21.5 HEU fuel bundles was utilized. The calculations were performed by positioning the Safety 1, Safety 2, and Regulating Blades at a critical position for the core, and then moving Safety Blade 3 through its full range of motion (2.5^{o} to 47.5^{o}). Results are provided in Table 4-14 and Figure 4-16. The total blade worth calculated here is 2.03% $\Delta k/k$, which is almost the same as the prior calculation for the total blade worth (2.06 % $\Delta k/k$). In the prior calculations, the other blades were fully-withdrawn, while in the calculations presented in Table 4-14, the safety blades were inserted at 38.5^{o} and the regulating blade was at 18.7^{o}.

Table 4-14 Integral Reactivity Worth versus Position for Safety 3 in the HEU Core

Time (s)[1]	**Blade Position**		**k_{eff} (depleted fuel)**	**Reactivity (%Δk/k)**	**Reactivity Insertion Rate (%Δk/k/s)**
	Degrees	**Units**			
0.0	2.5	0	0.98747 ± 0.04%	0.00%	n/a
5.6	5	56	0.98936 ± 0.03%	0.19%	0.034%
16.7	10	167	0.99330 ± 0.04%	0.59%	0.036%
27.8	15	278	0.99789 ± 0.04%	1.06%	**0.042%**
38.9	20	389	1.00158 ± 0.02%	1.43%	0.034%
50.0	25	500	1.00423 ± 0.02%	1.70%	0.024%
61.1	30	611	1.00576 ± 0.02%	1.85%	0.014%
72.2	35	722	1.00664 ± 0.02%	1.94%	0.008%
83.3	40	833	1.00704 ± 0.02%	1.98%	0.004%
100.0	47.5	1000	1.00747 ± 0.02%	2.03%	0.003%

[1] Assumes 100 seconds withdrawal time

Figure 4-16 shows calculated the Safety Blade 3 (most reactive blade for HEU core) worth as a function of position.

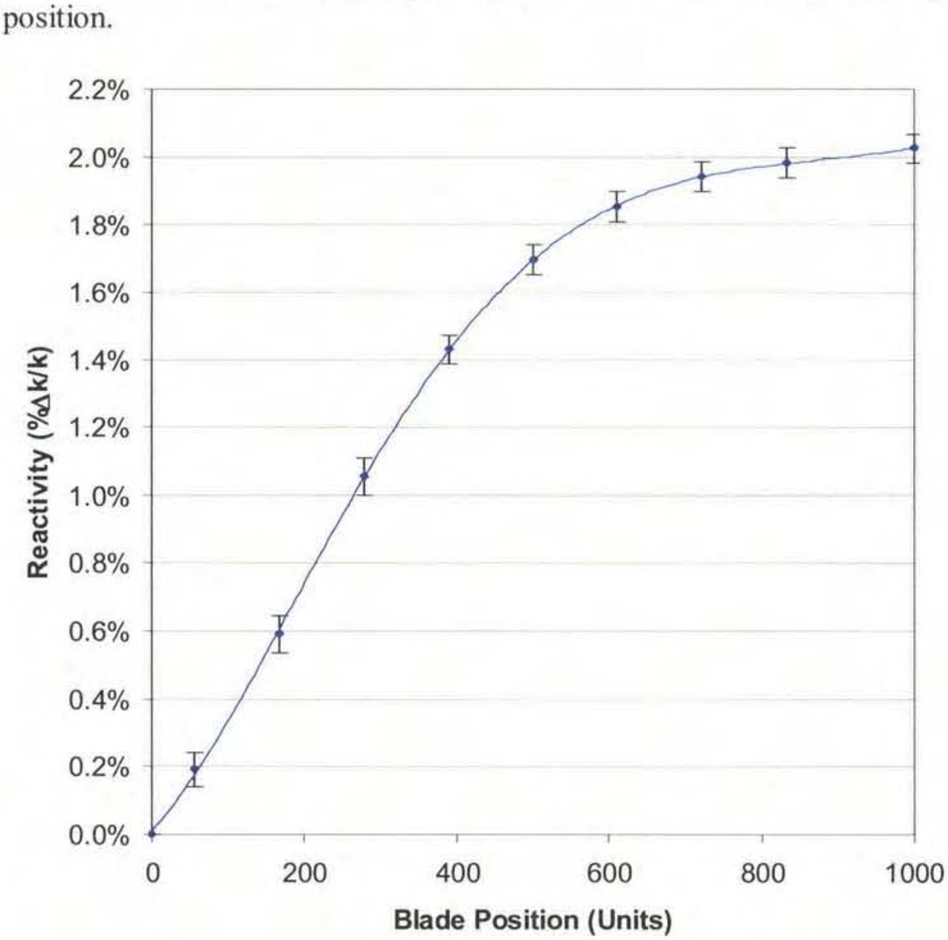


Figure 4-16 Integral Blade Worth versus Position for Safety 3 in the UFTR HEU Core.

The UFTR Technical Specifications require that the reactivity insertion rate from control blade withdrawal must be less than 0.06% $\Delta k/k/s$ when averaged over a 10 second interval. The rate of reactivity insertion resulting from withdrawal of the highest worth blade was approximated by assuming a 100 second (minimum allowed) blade withdrawal time. As shown in Table 4-14, the highest rate of reactivity insertion from withdrawal of Safety Blade 3 is 0.042% $\Delta k/k/s$, which meets the requisite UFTR Technical Specification.

<u>Maximum Reactivity Insertion Rate for LEU Core</u>

The integral reactivity worth as a function of position was determined for Safety Blade 2 (most reactive blade for the LEU core) based on MCNP calculations in a manner similar to that employed for the HEU core calculations. The position of Safety Blade 1 and Safety Blade 3 was fixed at 26.3° and the Regulating Blade was positioned at 16.9°, while Safety Blade 2 was rotated from 2.5° to 47.5°. Results are provided in Table 4-15 and Figure 4-17. The total worth for Safety Blade 2 calculated in this manner is similar to that obtained in the prior calculations with the other blades fully-withdrawn.

Table 4-15 Integral Reactivity Worth Versus Position for Safety 2 in the LEU Core

Time (s)[1]	Blade Position		k_{eff} (fresh fuel)	Reactivity (%Δk/k)	Reactivity Insertion Rate (%Δk/k/s)
	Degrees	Units			
0.0	2.5	0	0.98455 ± 0.04%	0.00%	n/a
5.6	5	56	0.98612 ± 0.03%	0.16%	0.029%
16.7	10	167	0.98954 ± 0.03%	0.51%	0.031%
27.8	15	278	0.99448 ± 0.04%	1.01%	**0.045%**
38.9	20	389	0.99701 ± 0.02%	1.27%	0.023%
50.0	25	500	0.99969 ± 0.02%	1.54%	0.024%
61.1	30	611	1.00114 ± 0.02%	1.69%	0.013%
72.2	35	722	1.00213 ± 0.02%	1.79%	0.009%
83.3	40	833	1.00238 ± 0.02%	1.81%	0.002%
100.0	47.5	1000	1.00229 ± 0.02%	1.80%	-0.001%

[1] Assumes 100 seconds withdrawal time

Figure 4-17 shows the calculated Safety Blade 2 (most reactive blade for LEU core) worth as a function of position.

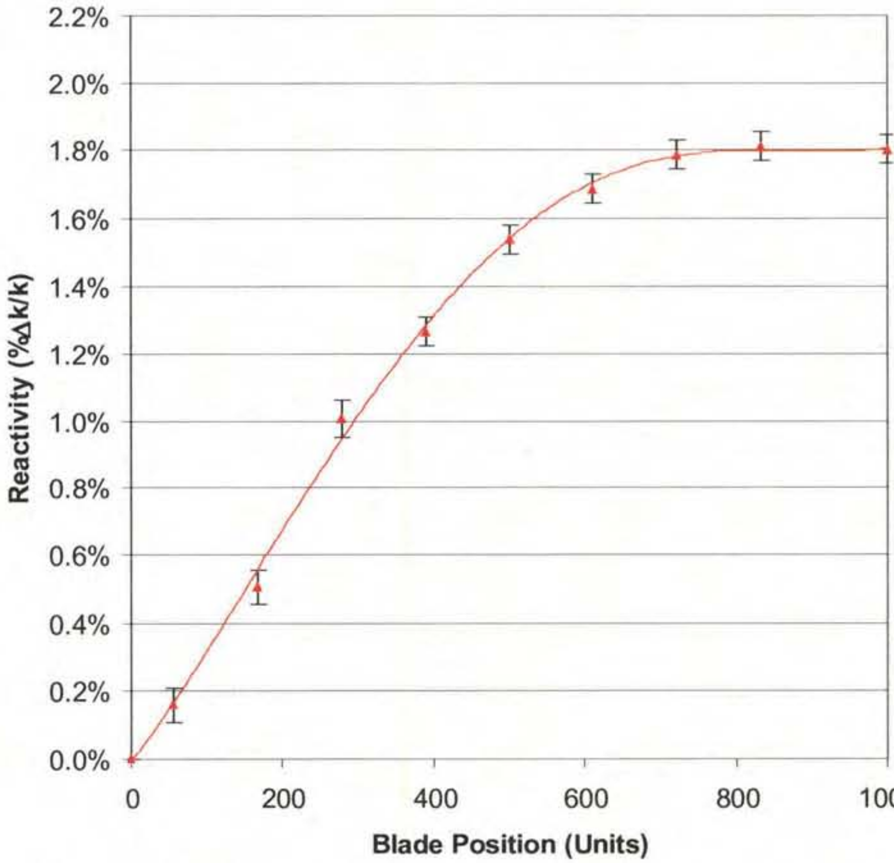


Figure 4-17 Integral Blade Worth versus Position for Safety 2 in the UFTR LEU Core

The UFTR Technical Specifications require that the reactivity insertion rate from control blade withdrawal must be less than 0.06% Δk/k/s when averaged over a 10 second interval. The rate of reactivity insertion resulting from withdrawal of the highest worth blade was again approximated by assuming a 100 second blade withdrawal time. As shown in Table 4-15, the highest rate of reactivity insertion from withdrawal of Safety 2 is 0.045% Δk/k/s, which also meets the Technical Specification for the reactor.

4.5.4 Shutdown Margin for HEU and LEU Cores

For the HEU core, the shutdown margin is evaluated by fully inserting (2.5 degrees from horizontal) the Safety Blades 1 and 2 and the Regulating Blade and withdrawing the Safety Blade 3 to its fully withdrawn position (47.5 degrees from horizontal). For the LEU core, the Safety Blade 2 is kept fully withdrawn while the others blades are fully inserted. Table 4-16 compares the shutdown margins of the HEU and LEU cores.

Table 4-16 Shutdown Margins for the Current HEU Core and the Reference LEU Core

	Depleted HEU Core (calculated)	HEU Core (measured)	Reference LEU Core
Shutdown Margin (Δk/k %)	3.11	3.01	3.17

Since the measured shutdown margin for the HEU depleted core is 3.01% the calculated and experimental shutdown margins differ by about 3%. The LEU shutdown margins meet the Technical Specification requirement that the shutdown margin be at least 2% Δk/k with the most reactive blade stuck out.

4.5.5 Other Core Physics Parameters

HEU Reactivity Coefficients and Kinetic Parameters

Reactivity coefficients and neutron kinetics parameters were calculated for the HEU-fueled UFTR. These provide a measure of the core reactivity response to changes in the water properties or fuel temperature changes under both off-nominal (e.g., changes to inlet coolant conditions) and accident conditions (e.g., inadvertent reactivity insertion accidents). The Technical Specifications for the UFTR require that the primary coolant temperature and void coefficients be negative.

The reactivity coefficients are used to estimate the core reactivity change due to a change in some state property value. So,

$$\Delta\rho = \alpha_x \cdot \Delta x \tag{4.1}$$

where α_x is the reactivity coefficient due to a unit change in state property x and Δ_x is the value change for x. The reactivity coefficients are calculated assuming that reactivity effects resulting from simultaneous changes in multiple state properties are separable. These are calculated from core eigenvalue calculations with independent perturbations to the state properties. Consequently,

$$\alpha_x = \frac{\Delta\rho}{\Delta x} = \frac{k_1 - k_o}{k_1 k_o} \cdot \frac{1}{(x_1 - x_o)}. \tag{4.2}$$

For the UFTR, reactivity coefficients were calculated for perturbations to the water temperature, water density (coolant void), and fuel temperature. Core eigenvalue calculations were performed with the MCNP5 code using the same core model that was used to evaluate the steady-state neutron flux distribution, excess reactivity, and control blade reactivity worth.

The kinetics parameters evaluated for the UFTR were the effective delayed neutron fraction, β_{eff}, and the prompt neutron lifetime, ℓ. The effective delayed neutron fraction is calculated using two eigenvalue calculations. The normal calculation of k_{eff} will include both prompt and delayed neutrons. A second calculation with prompt neutrons only yielded from fission gives k_{eff}^{prompt}. The effective delayed neutron fraction is then defined as

$$\beta_{eff} = 1 - \frac{k_{eff}^{prompt}}{k_{eff}}. \tag{4.3}$$

The prompt-neutron lifetime is calculated using the "1/v insertion method," in which a uniform concentration of a 1/v absorber such as B^{10} is included at a very dilute concentration everywhere in the core and reflector. Consequently, the lifetime is calculated by the formulation

$$l = \lim_{N\to 0}\left[\frac{\dfrac{k_1 - k_o}{k_1}}{N_1 \sigma_a \mathrm{v}}\right], \tag{4.4}$$

where k_1 is the k_{eff} of the system with a uniform concentration, N_1, of a 1/v absorber, and σ_a is the infinitely-dilute absorption cross section of the absorber for neutrons at speed v. In this work, the ^{10}B absorption cross section was assumed to be $\sigma_a = 3837$ barns at a neutron speed of v = 2200 m/s.

Table 4-17 provides the reactivity coefficients and kinetics parameters calculated for the UFTR HEU core. The calculations were performed for the fresh HEU core with 21.5 bundles. This core has an excess reactivity of 1.09% $\Delta k/k$, and the control blades were positioned to achieve a critical condition. The reactivity coefficients are all negative, which meets the requirements of the Technical Specifications. The fuel temperature (Doppler) coefficient is quite small for the HEU fuel (93 wt.% U-235) because of the very small fraction of U-238.

Table 4-17 Kinetics Parameters and Reactivity Coefficients for the UFTR HEU Core

Parameter		Calculated Result
β_{eff}		7.92E-03 ± 1%
ℓ (μs)		187.4 ± 3%
α_{void} (Δρ/%void)	*(0 to 5% void)*	-1.48E-03 ± 1%
	(5 to 10% void)	-1.69E-03 ± 1%
α_{water} (Δρ/°C)	*(21 to 127°C)*	-5.91E-05 ± 1%
α_{fuel} (Δρ/°C)	*(21 to 127°C)*	-6.49E-06 ± 18%
	(21 to 227°C)	-2.91E-06 ± 12%

The ranges on coolant voiding (change of density) and temperature selected here cover perturbations that will occur during normal operations. The calculated coefficients show that there is some non-linearity in the reactivity response to coolant voiding and temperature, and fuel temperature changes. When performing coupled thermal-hydraulics/neutronics analyses for transients, the expected range of the coolant and fuel conditions should be taken into account when selecting the coefficients to employ. The RELAP5-3D code (Ref. 10) which was used for transient analyses does not easily allow the input of temperature dependent coefficients, so a single value must be selected.

<u>LEU Reactivity Coefficients and Kinetic Parameters</u>
Kinetics parameters and reactivity coefficients were calculated for the reference LEU core at fresh and depleted conditions, as noted in Section 4.5.3. The excess reactivity for the fresh LEU core is 0.93% Δk/k, and the control blades were positioned to achieve a critical core condition when calculating the reactivity coefficients. The depleted core conditions were calculated assuming 20 years of reactor operations at 20 hours/week. The calculated reactivity of the core after this irradiation history is sub-critical (-0.42% Δk/k), even with the control blades completely withdrawn. It is judged that the fresh and depleted cores cover the range of conditions affecting the accident response of the LEU-fueled UFTR. The calculated parameters are summarized in Table 4-18.

Table 4-18 Kinetics Parameters and Reactivity Coefficients for the UFTR LEU Core.

Parameter		Fresh Core	Depleted Core
β_{eff}		0.00771 ± 1%	0.00756 ± 2%
ℓ (μs)		177.5 ± 5%	195.1 ± 6%
C_{void} (Δρ/%void)	*(0 to 5% void)*	-1.53E-03 ± 1%	-1.46E-03 ± 2%
	(5 to 10% void)	-1.75E-03 ± 1%	-1.65E-03 ± 2%
C_{water} (Δρ/°C)	*(21 to 127°C)*	-5.68E-05 ± 2%	-5.26E-05 ± 3%
C_{fuel} (Δρ/°C)	*(21 to 127°C)*	-1.76E-05 ± 6%	-1.72E-05 ± 9%
	(21 to 227°C)	-1.65E-05 ± 3%	-1.49E-05 ± 4%

The LEU fuel has a much larger fuel temperature (Doppler) coefficient relative to the HEU-fueled UFTR because of the higher U-238 content of the fuel. The fresh LEU core has a harder neutron spectrum than the HEU core, which slightly decreases the prompt-neutron lifetime, increases the magnitude of the coolant void coefficient, and slightly reduces the magnitude of the coolant temperature coefficient. All coefficients are negative, as required by the Technical Specifications.

The neutron spectrum softens with depletion of the LEU fuel in the UFTR. Consequently, the prompt-neutron lifetime becomes longer and the magnitude of the coolant void coefficient is reduced as the core is depleted.

The ranges on coolant voiding and temperature selected here cover any perturbations that will occur during normal operations. Just as for the HEU fuel, the calculated coefficients show that there is some non-linearity in the reactivity response to coolant voiding and temperature, and fuel temperature changes. When performing coupled thermal-hydraulics/neutronics analyses for transients, the expected range of the coolant and fuel conditions should be taken into account when selecting the coefficients given in Table 4-18 above.

4.6 Functional Design of the Reactivity Control System

The proposed HEU to LEU conversion of the UFTR core does not require any changes in the functional design of the reactivity control system. More details about this topic can be found in Ref. 1.

4.7 Thermal-hydraulic Analyses

In this section, the results of thermal-hydraulic analyses are discussed in order to demonstrate that the UFTR thermal-hydraulic LEU design provides the cooling conditions necessary to ensure fuel integrity under all anticipated reactor operating conditions. Analyses for operation under accident scenarios are presented in Section 13.

4.7.1 Fuel Assembly and Fuel Box Geometry

In the UFTR, fuel is loaded into six "fuel boxes," each containing up to four fuel assemblies. Diagrams comparing the relative sizes of the HEU and LEU fuel assemblies are shown in Figure 4-18. The bolt-to-bolt, or stack height, dimension for the LEU assembly was designed to be smaller than that for the HEU assemblies so that the largest LEU fuel assemblies allowed by the manufacturing tolerances will fit more easily into the smallest fuel box than did the HEU assemblies. The arrangement of four fuel assemblies inside a fuel box is shown in Figure 4-19.

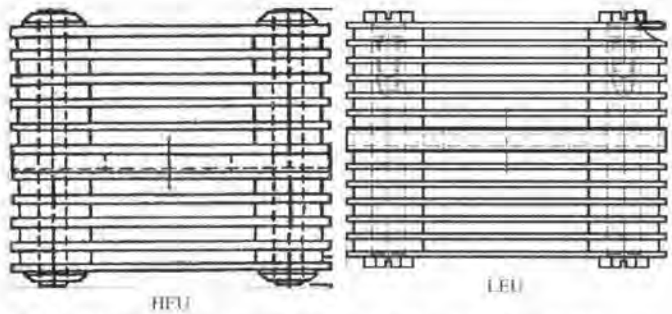


Figure 4-18 Diagrams Comparing the LEU and HEU assemblies

[redacted]

Figure 4-19 Arrangements of the HEU and LEU Fuel Assemblies in a Fuel Box

The HEU assembly configuration uses a central tapered wedge pin to force the four assemblies to the corners of the fuel box, as shown in Figure 4-19. The analysis assumes that the assemblies are in contact with the fuel box. This leaves a wide East-West channel in the center of the fuel box. If four nominally-dimensioned HEU assemblies allowed by the manufacturing tolerances are placed in the largest fuel box, the central East-West channel is 0.571" (14.5 mm) wide (bolt heads in contact with the fuel box).

The LEU design uses two wedge pins to position the fuel assemblies in each fuel box, as shown in Figure 4-19. The two-pin LEU configuration with the smallest assemblies in the largest box produces two wide East-West channels of width 0.3255" (8.27 mm). Had the single pin HEU design been employed here, there would have been one large central East-West channel of width 0.651" (16.5 mm). This very wide channel would have consumed a disproportionate amount of the total coolant flow, leaving less to cool the other fuel plates in the fuel box. Therefore, the two-pin configuration is hydraulically superior to the single pin configuration in that it causes more flow into the narrower coolant channels where it is needed most. In both the HEU and the two-pin LEU designs the 0.435" central North-South channel is maintained.

In the thermal-hydraulic analyses, all four fuel assemblies and the interior volume of the fuel box with the limiting power density distribution were modeled explicitly using the PLTEMP code (Ref. 11). In the HEU core, this fuel box contained fuel in locations 5-1, 5-2, 5-3, and 5-4. The limiting fuel assembly was in location 5-1. In the LEU core, the limiting fuel box contained assemblies in locations 2-1, 2-2, 2-3, and 2-4. The limiting fuel assembly was in location 2-3. The relative power densities in each fuel plate were obtained from detailed MCNP5 criticality calculations. In the PLTEMP analysis, the relative axial power profile of the hottest fuel plate of the hottest fuel assembly was applied to all fuel plates.

Hot channel factors are used to account for dimensional variations inherent in the manufacturing process, as well as variations in other parameters that affect thermal-hydraulic performance. The dimensions that were used in the HEU and LEU thermal-hydraulics models are shown in Table 4-19.

Table 4-19 Key Geometrical Parameters Used in the HEU and LEU Models

Model Geometric Parameter	HEU		LEU	
	inches	Mm	inches	Mm
Fuel box interior depth	[redacted]			
Fuel box interior width				
Fuel plate thickness	0.070	1.78	0.050	1.27
Channel thickness against fuel box	0.137	3.48	0.3255	8.268
Central horizontal channel thickness	0.571	14.5	0.188	4.78
Vertical bypass gap thickness	0.435	11.0	0.435	11.0
Coolant channel thickness	0.137	3.48	0.111	2.82
Bolt head height	0.137	3.48	0.094	2.39

The grid plate, which supports the four fuel assemblies in each fuel box, is included in the hydraulic analysis because it makes the velocity distribution in each fuel box more uniform. The hydraulic model in the code assumes that the hydraulic resistance for each coolant path, from the bottom of the grid plate to the region above the fuel plates, has two components, a form- or k-loss and a frictional loss. For each of these parallel paths or channels the pressure drop, ΔP, is given by $\Delta P = (K + fL/D) \times \rho V^2 / 2$, where K is the k-loss value, f is the friction factor for smooth-walled channels, L is the channel length, D is the channel hydraulic diameter, ρ is the coolant density, and V is the average coolant velocity in the channel. For laminar flow the value of f is affected by the shape of the channel. The single value of K represents not only the form losses at the inlet and exit to the fuel plates, but also the hydraulic resistance due to the grid plate. The minimum total flow area in the grid plate is considerably smaller than the total flow area in the fuel region. Also, there are multiple parallel flow paths through each fuel box that were considered in this hydraulic analysis. For each path the flow passes first through the grid plate and then through the fuel assembly region. As explained in Appendix A.8, for each of these parallel flow paths, the value of 5 that was used in the analyses is a conservatively low value for the effective K-loss for each path. The appendix also provides a parametric study that covers all possible values of K from 0 to infinity and indicates the sensitivity of the thermal results to the choice of K. A value of 5.0 was assumed for the value of K and is considered to be conservatively small. A larger value of K would result in larger margins to the limiting conditions, such as the onset of nucleate boiling, by causing the thinner channels to have more flow. Since the ends of the side edges of the fuel plates are open where they abut the side channel, in theory there can be some flow between the fueled channels and the side channel through the center of the fuel box. However, in general, this lateral flow is expected to be small since the local pressure is expected to be essentially uniform at each axial level. The higher vertical flow velocities in the bigger channels, which have the larger hydraulic diameters, tend to keep the axial pressure drops through each of the parallel paths equal and the pressures uniform at each axial level. When the pressure is uniform at each axial level, there is no mechanism for redistribution of flow among adjacent open channels. Thus, any impact of any flow diversion should be small. Moreover, the hot channel factors include a 20% uncertainty in channel flow distribution as a random error. Additional information is provided in Appendix A.8.

4.7.2 PLTEMP/ANL v3.0 Code Description

Thermal-hydraulic analyses were performed using the computer code PLTEMP/ANL V 3.0 (Ref. 11). This code provides a steady-state thermal-hydraulics solution for research reactor fuel assemblies with plate-type or tube-type geometries. The code accounts for pressure drops axially in one dimension including any bypass flows, and accounts for thermal effects in two dimensions. The third dimension is along the width of the plate. Width effects such as heated area not being the same as wetted area are accounted for. The coolant channel hydraulic diameter, area, and friction factor are obtained assuming that the fuel plates are in contact with the sides of the fuel box, and the channel is the full width of the fuel plate. Friction factors and mass flow rates are determined through a network of parallel channels, some of which are not heated. Both laminar and turbulent flow regimes are accommodated by PLTEMP, although the UFTR operates in the laminar flow regime.

The heat source from fission is assumed to be flat across the meat and along the width of a fuel plate, but varies axially in a step-wise nodal approximation. A computational fluid dynamics (CFD) analysis with the STAR-CD code (Ref. 28) was performed for a single LEU fuel plate with and without a power profile along the width of the fuel (see Appendix A.9). The results of the CFD calculations showed that the conductivity of the fuel plate is sufficient to flatten the temperature profile along the width of the fuel plate. The CFD calculations showed that the flat profile assumption in PLTEMP for the power profile across the width needs no correction for determination of safety parameters. The peak heat flux computed by PLTEMP is conservative in that it is based on the heated perimeter, while the CFD analysis shows that the power is distributed across the full plate width.

PLTEMP determines the friction factors and coolant mass flow rates in each channel, and then calculates the steady-state temperature distribution in the meat, clad, and coolant at each axial node. The computational process begins at the inlet end of the channel, and proceeds level by level to the channel outlet.

The code accounts for one-sided heating of a channel, as occurs for the channel next to the fuel box. In laminar flow, the heat transfer coefficient is different for a channel heated on one side than for a channel heated on two sides. Also, Version 3.0 accounts for pressure drop friction factors over the full Reynolds number range from laminar, through the critical zone, and on through turbulent flow.

Safety-related parameters such as the Onset of Nucleate Boiling Ratio (ONBR) and Departure from Nucleate Boiling Ratio (DNBR) are calculated along with fuel, clad, and coolant temperatures in each channel.

The major thermal-hydraulic correlation options in PLTEMP that were selected for use in these analyses are:

1. Nu=hD/k=7.63, if laminar forced convection, or Nu=4.86, if laminar forced convection and the channel is heated on one side only (Ref. 12). Note: turbulent flow does not take place. If it did, the Petukhov & Popov (Ref. 13) single-phase heat transfer correlation would be used.
2. Bergles-Rohsenow correlation (Ref. 14) for ONB thermal margin.
3. Groeneveld Lookup Table for Critical Heat Flux Prediction (Ref. 15).

The hot channel factors that were used in the HEU and LEU cores are shown in Table 4-20 and Table 4-21, respectively. The methodology for determining the hot channel factors and for applying them in PLTEMP are described in Appendix A.10.

It is worth noting that the systematic uncertainties in the measurements of the reactor power level and the coolant flow rate were not included explicitly in the calculations. These systematic uncertainties will be included in the interpretation of the results for the limiting safety system settings in Section 4.7.4.

Table 4-20 Hot Channel Factors for the HEU Core

						Hot Channel Factors			
uncertainty	**type of tolerance**	**effect on bulk ΔT, fraction**	**value**	**toler-ance**	**toler-ance, fraction**	**heat flux, F_q**	**Chanel flow rate, F_w**	**channel temper-ature rise, F_{bulk}**	**film temper-ature rise, F_{film}**
fuel meat thickness (local)[a]	random				0.05	1.05			1.05
^{235}U homogeneity (local)[b]	random				0.03	1.03			1.03
^{235}U loading per plate[c]	random	0.50			0.03	1.03		1.015	1.03
power density[a]	random	0.50			0.10	1.10		1.05	1.10
channel spacing	random	1.00	0.137	0.001	1.007		1.022	1.022	1.007
flow distribution	random	1.00			0.20		1.20	1.20	1.00
random errors combined						**1.12**	**1.20**	**1.21**	**1.12**
power measurement	systematic	1.00			**0.00[d]**	1.00		1.00	1.00
flow measurement	systematic	1.00			**0.00[d]**		1.00	1.00	1.00
heat transfer coefficient	systematic				**0.20**				1.20
systematic errors combined						1.00	1.00	1.00	1.20
product of random & systematic						**1.12**	**1.20**	**1.21**	**1.34**

[a] Assumed values. [b] Estimated for U-Al alloy fuel meat. [c] Assumed to be the same 3% loading uncertainty as for the LEU fuel plate. [d] These factors have been set to zero because true power and true flow rates are used in the PLTEMP calculations.

Table 4-21 Hot Channel Factors for the LEU Core *

						Hot Channel Factors			
uncertainty	**type of tolerance**	**effect on bulk ΔT, fraction**	**Value**	**toler-ance**	**toler-ance, fraction**	**heat flux, F_q**	**channel flow rate, F_w**	**channel temper-ature rise, F_{bulk}**	**film temper-ature rise, F_{film}**
fuel meat thickness (local)	random				0.00	1.00			1.00
^{235}U homogeneity (local)[a]	random				0.20	1.20			1.20
^{235}U loading per plate[b]	random	0.50			0.03	1.03		1.015	1.03
power density[c]	random	0.50			0.10	1.10		1.050	1.10
channel spacing, in. or mm	random	1.00	0.111	0.001	1.009		1.028	1.028	1.009
flow distribution	random	1.00			0.20		1.20	1.20	1.00
random errors combined						**1.23**	**1.20**	**1.21**	**1.23**
power measurement	systematic	1.00			**0.00[d]**	1.00		1.00	1.00
flow measurement	systematic	1.00			**0.00[d]**		1.00	1.00	1.00
heat transfer coefficient	systematic				**0.20**				1.20
systematic errors combined						**1.00**	**1.00**	**1.00**	**1.20**
product of random & systematic						**1.23**	**1.20**	**1.21**	**1.47**

[a] Derived from fuel plate loading specification of [redacted]5 ± [redacted] g ^{235}U. [b] From fuel plate homogeneity specification.
[c] Assumed value. [d] These factors have been set to zero because true power and true flow rates are used in the PLTEMP calculations. The effects of these two systematic uncertainties are included in the interpretation of the results.

***With 1 mil tolerance for the water channel spacing**

4.7.3 Thermal-Hydraulic Analysis Results

In this section, the PLTEMP/ANL V 3.0 code (Ref. 11) is used to determine the thermal-hydraulics parameters of the UFTR under nominal conditions for both HEU and LEU cores. The code is then used to compute the value of reactor power at which onset of nucleate boiling is reached for various coolant flow rates and three parametric values of the reactor coolant inlet temperature. These power versus flow rate relationships are used to select the Limiting Safety System Settings (LSSS) (or trip points) and to define the allowed operating region for the LEU core.

Nominal Operating Conditions

The nominal operating conditions for the HEU and LEU cores are listed in Table 4-22.

Table 4-22 Nominal Operating Conditions for the HEU and LEU Cores

	Nominal Condition
Inlet Temperature, C	30 (86 F)
Inlet mass flow rate, kg/s	2.688 (43 gpm)*
Power, kW	100

*With 1 mil tolerance for the water channel spacing

Table 4-23 compares the thermal-hydraulics parameters of the HEU and LEU cores at nominal operating conditions. All hot channel factors are included in the calculations, except for uncertainties in measurements of the power level, coolant flow rate, and inlet temperature. For both HEU and LEU cores, the maximum fuel temperature and the maximum clad temperatures occurred at a height of 57.5 cm from the bottom of the fuel meat.

Table 4-23 Thermal-hydraulics Parameters of the HEU and LEU Cores at Nominal Operating Conditions

Parameter	HEU	LEU
Max. Fuel Temp., C	66.5	64.5
Max. Clad Temp., C	66.5	64.4
Mixed Mean Coolant, outlet temperature, C	40.8	40.5
Max. Coolant Channel, outlet temperature, C	58.3	59.1
Min. ONBR	1.98	2.09
Min. DNBR	354	376

The minimum ratios for Onset of Nucleate Boiling are calculated to be 1.98 in the HEU core and 2.09 in the LEU core. The minimum ratios for Departure from Nucleate Boiling (DNB) are calculated to be 354 and 376 in the HEU and LEU cores, respectively. Thus, both the HEU and LEU cores have adequate thermal-hydraulic safety margins under normal operating conditions.

Safety Limits for the LEU Core

In the UFTR, the first and principal physical barrier protecting against release of radioactivity is the cladding of the fuel plates. The 6061 aluminum alloy cladding has an incipient melting temperature of 582 °C. However, measurements (NUREG-1313, Ref. 2) on irradiated fuel plates have shown that fission products are first released near the blister temperature (~550 °C) of the cladding. To ensure that the blister temperature is never reached, NUREG-1537 (Ref. 16) concludes that 530 °C is an acceptable fuel and cladding temperature limit not to be exceeded under any conditions of operation. As a result, the UFTR has proposed a safety limit in its Technical Specifications requiring that the fuel and cladding temperatures should not exceed 530 °C.

In sections 4.7.3.1 and 4.7.3.2, we present analyses performed for different water channel spacing tolerance for both HEU and LEU cores. Section 4.7.3.1 gives the results for a 1 mil tolerance, while section 4.7.3.2 gives the results for 10,15, and 20 mil tolerance for the HE?U core. Section 4.7.4 provides important thermal parameters for the LEU core

4.7.3.1 Limiting Safety System Settings for the LEU Core (with water channel spacing tolerance of 1 mil)

Limiting safety system settings (LSSS) for nuclear reactors are settings for automatic protective devices related to those variables having significant safety functions. When a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a safety limit is reached. This section provides the methodology and rationale for selecting the LSSS to define the operating region for the UFTR LEU core, taking into account uncertainties in measurements of the reactor power, coolant flow rate, and average primary inlet coolant temperature.

For steady-state operation, peak clad temperatures are maintained far below 530 °C by avoiding flow instability. Flow instability, in turn, is avoided by ensuring that onset of nucleate boiling does not occur. The data that was used for selection of the LSSS for the UFTR LEU core is shown in Table 4-24. This table shows calculated values of the true reactor power at which onset of nucleate boiling (ONBR) occurs for different values of the true coolant flow rate and three values of the average primary coolant inlet temperature. These data are plotted in Figure 4-20.

Table 4-24 Data for Selection of Limiting Safety System Settings: Calculated Reactor Power at which ONBR = 1.0 for Several Coolant Flow Rates and Three Values of the Inlet Temperature*(with water channel space tolerance of 1 mil).

Tin, F Tin, C	86 30				100 37.8				110 43.3			
Flow Rate, gpm	18	30	43	50	18	30	43	50	18	30	43	50
Power, kW	91.4	156.9	225.1	259.8	81.3	140.2	201.3	232.5	74.3	128.3	184.3	213.0
T_{fuel}, Max, C	101.1	101.4	101.7	101.8	101.1	101.4	102.1	102.3	101.1	101.4	101.6	102.2
T_{clad}, Max, C	101.1	101.4	101.6	101.7	101.1	101.3	102.1	102.2	101.1	101.4	101.5	102.2
T_{out}, Max, C	97.3	93.5	90.1	88.3	97.8	94.2	91.7	90.1	98.1	94.8	92.1	91.0
T_{out}, Mixed Mean, C	53.0	53.7	53.7	53.5	58.3	59.0	59.0	58.9	62.1	62.7	62.8	62.7
Min ONBR	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
Min DNBR	221	157	126	115	232	165	132	120	241	171	136	125

* In coupled neutron-gamma transport calculations using MCNP, 95% of the total power generated was calculated to be deposited in the fuel meat. A conservative value of 2% of the power generated was assumed to be deposited directly into the coolant water.

Figure 4-20 also shows horizontal and vertical solid lines that represent the true maximum power level of 125 kW and the true minimum coolant flow rate of 34 gpm. The true maximum value of the average coolant inlet temperature is 110 °F.

The limiting safety system settings for the LEU core were chosen to be:

(1) The power level at any flow rate shall not exceed 119 kW.
(2) The primary coolant flow rate shall be greater than 36 gpm at all power levels greater than 1 W.
(3) The average primary coolant inlet temperature shall not exceed 109 °F.
(4) The average primary coolant outlet temperature shall not exceed 155 °F.

This selection of LSSS values restricts operation of the reactor to that portion of Figure 4-20 labeled "Operating Region", taking into account uncertainties in measurements of reactor power (±5%), coolant flow rate (±5%), average primary coolant inlet temperature (± 1 °F), and average primary coolant outlet temperature (± 1 °F).

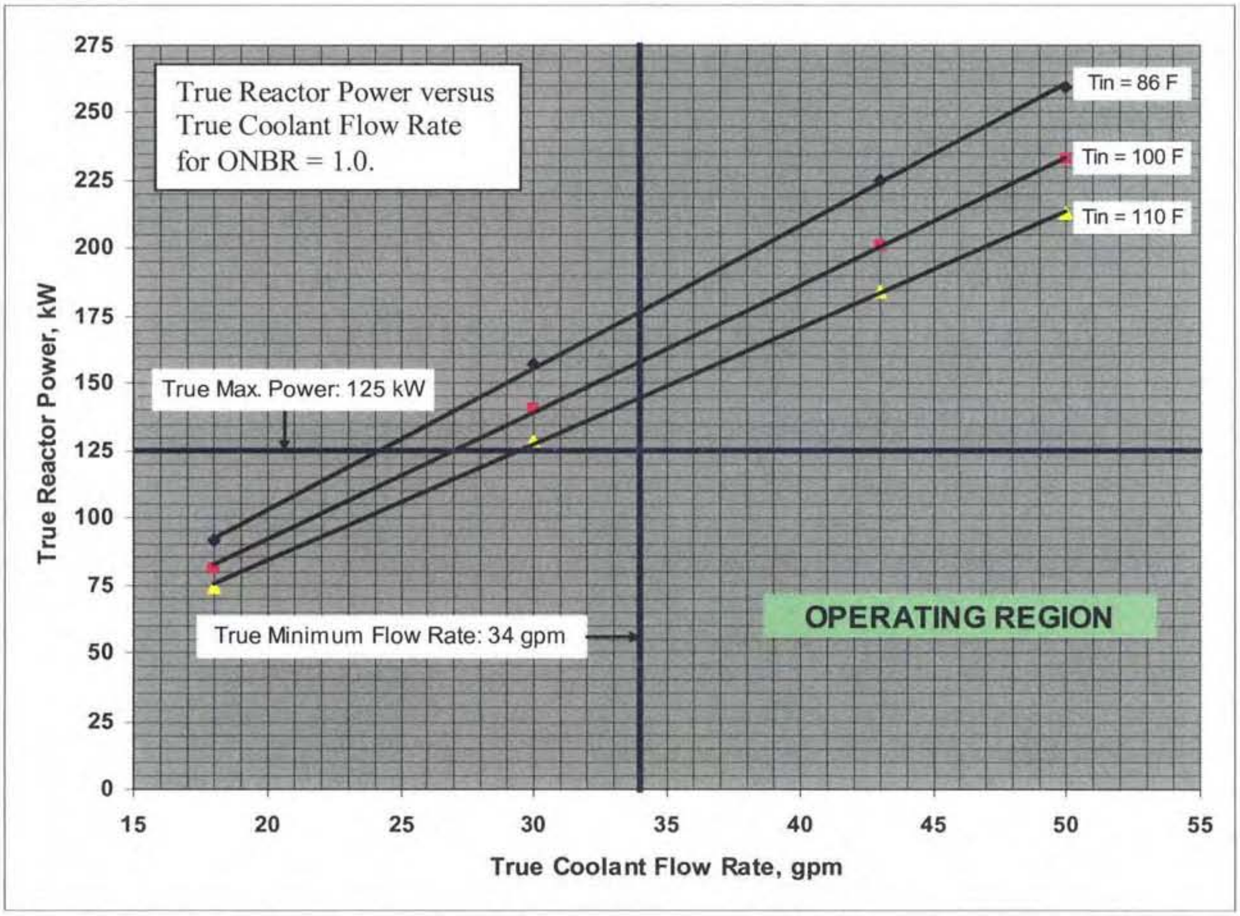


Figure 4-20 Calculated Values of True Reactor Power at Different True Coolant Flow Rates for ONBR = 1.0 at Three Inlet Temperatures (with 1 mil tolerance for water channel spacing).

The LSSS that were selected are thus conservative settings that pre-empt the possibility of a premature burnout of the core and possible damage to the fuel plates due to a flow instability during steady-state operation. The data in Table 4-24 also show that there are very large margins to departure from nucleate boiling (DNBR). Consequently, the selected LSSS are conservative settings that protect the reactor by preventing the temperature of the fuel and cladding from reaching the safety limit of 530 °C during steady-state operation.

Chapter 13 (Accident Analyses) analyzes two hypothetical transients based on values proposed in the Technical Specifications for the LEU core. These transients are (1) Rapid insertion of the maximum reactivity worth of 0.6% Δk/k of all moveable and non-secured experiments and (2) Slow insertion of reactivity at the maximum allowed rate of 0.06% Δk/k/second due to control blade withdrawal. In each case, the analyses assumed that the first LSSS that was reached for a monitored reactor parameter failed to initiate reactor scram. Scram was initiated, however, in response to the second reactor parameter reaching an LSSS setting.

For the case of a rapid insertion of 0.6% Δk/k, the reactor period trip was assumed to fail. The reactor protection system initiated a reactor scram based on the power level LSSS. The temperature of the fuel and cladding were calculated to increase by about 1 °C above the maximum steady-state value. The maximum fuel and cladding temperature is far below the safety limit of 530 °C.

For the case of a slow insertion of 0.06% Δk/k/second, one of the two power measurement channels of the reactor protection system was assumed to fail. Reactor scram was initiated by the redundant LSSS on reactor power. The temperature of the fuel and cladding were calculated to increase by about 1 °C above the maximum steady-state value. The maximum fuel and cladding temperature is far below the safety limit of 530 °C.

Thus, the LSSS that were selected are also conservative settings which ensure that the maximum fuel and cladding temperatures do not reach the safety limit of 530 °C for the range of accident scenarios that were analyzed.

4.7.3.2 Effects of Increased Water Channel Spacing Tolerance

Initial inspection and measurement of LEU fuel assemblies manufactured by BWXT in July 2006 showed a variation in the water channel spacing that was larger than expected. In this design, the ends of the plates are separated by aluminum spacers and are bolted together. Aluminum spacers are welded onto the edges of the plates at about half their height. To eliminate the observed variations of water channel spacing, this fuel assembly design was modified by BWXT to include the "comb" design shown in Figure 4-21 that will physically separate the fuel plates at the nominal quarter-points along the fuel plate length. In this modified design, the tolerance on the minimum water channel spacing is expected to be a maximum of ± 20 mils. The nominal water channel spacing at the bolted ends of the fuel assembly on the manufacturing drawings is 110 – 112 mils, giving a minimum water channel spacing of 90 mils.

[redacted]

Figure 4-21 Modified Fuel Assembly Design by BWXT to include the "Comb" Design

The actual minimum water channel spacing of each water channel in each finished fuel assembly will be measured by BWXT and made available by August 10, 2006. In case any one of the channel spacings is less than 90 mils, another comb will be added to maintain a minimum channel spacing of 90 mils.

Thermal-hydraulic analyses are provided in this section assuming tolerances on the water channel spacing of 10, 15, and 20 mils, along with repositioning of the fuel assemblies in the fuel boxes due to the 65 mil protrusion of the combs beyond the ends of the fuel plates. The same hot-channel-factor methodology described in prior analysis was used to compute curves of true reactor power versus the true coolant flow rate at which onset of nucleate boiling (ONB) would occur for several values of the average coolant inlet temperature.

The three key parameters to be examined are the true minimum coolant flow rate, the Limiting Safety System Setting (LSSS) for the coolant flow rate, and the normal operating coolant flow rate.

New Hot Channel Factors and Revised Model

The steady-state thermal hydraulic analysis described in prior analysis assumed hot channel factors based on a 1 mil tolerance for the water channel spacing, as shown on the manufacturing drawings at the bolted ends of the fuel assemblies. In the model, the ends of the fuel plates were assumed to be in contact with the fuel box wall. These new analyses account for two effects – the larger tolerance on the water channel spacing described above and the repositioning of the fuel assemblies in the fuel boxes due to the thickness of the combs extending beyond the edges of the fuel plates. The combs introduce into the computer model another bypass channel pathway that is located between the fuel plate ends and the box wall. In these new analyses, the ends of the fuel plates are moved away from the wall of the fuel box by 0.065 inches - the height of the comb above the "teeth", as shown in Figure 1.

However, the newly created slot is actually blocked by the width of the comb (0.190 inches), at the quarter-points along the length of the assembly. There will be a small bypass flow through this pathway that is conservatively accounted for by including it as a clear channel, ignoring the blockage created by the presence of the combs. The pressure drop introduced by the combs is not accounted for because it is very small. As a result, the computed coolant flow through the heated channels will be reduced, and the ONB margin and other parameters will be conservative. The presence of the comb also reduces the width of the central channel slot between fuel assemblies. This is a significant beneficial change because it reduces the bypass flow through the central slot.

The hot channel factors shown in Table 4-25 were computed for coolant channel spacing tolerances of 10, 15, and 20 mils using the prior methodology described and related appendices.

Table 4-25 Hot Channel Factors as a Function of Tolerance on Coolant Channel Width

Hot Channel Factor	**1 mil**	**10 mil**	**15 mil**	**20 mil**
F_{bulk}	1.21	1.39	1.58	1.84
F_{film}	1.23	1.24	1.26	1.29

Calculated Results

Figure 4-21 shows the curves from the prior analysis of true reactor power versus true coolant flow rate at which ONB occurs for 1 mil water channel spacing case. The key parameters are shown for reference purposes in Table 4-26

Table 4-26 Key Parameters for Reactor Power and Coolant Flow Rate.

Parameter	**Value**	**Parameter**	**Value**
True Maximum Power	125 kW	True Minimum Flow Rate	34 gpm
LSSS Power	119 kW	LSSS Flow Rate	36 gpm
Maximum Operating Power	100 kW	Operating Flow Rate	43 gpm

The new power versus coolant flow rate curves for water channel spacing tolerances of 10 mils, 15 mils, and 20 mils, along with repositioning of the fuel assemblies in the fuel box to account for the thickness of the combs are shown in Figures 4-22, 4-23, and 4-24, respectively.

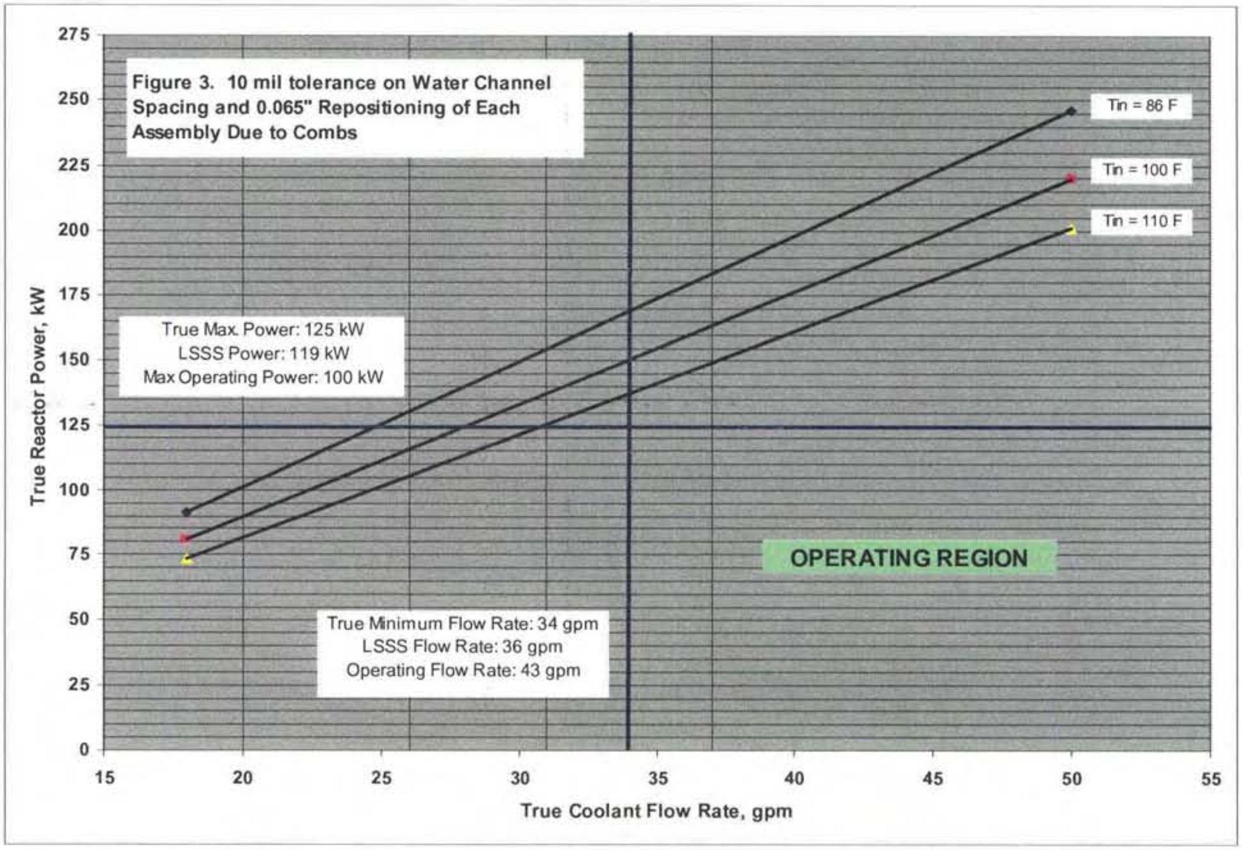


Figure 4-22 True power versus true flow rate at which ONB Occurs (with 10 mil tolerance on water channel spacing and repositioning of assemblies in fuel box to account for thickness of combs.)

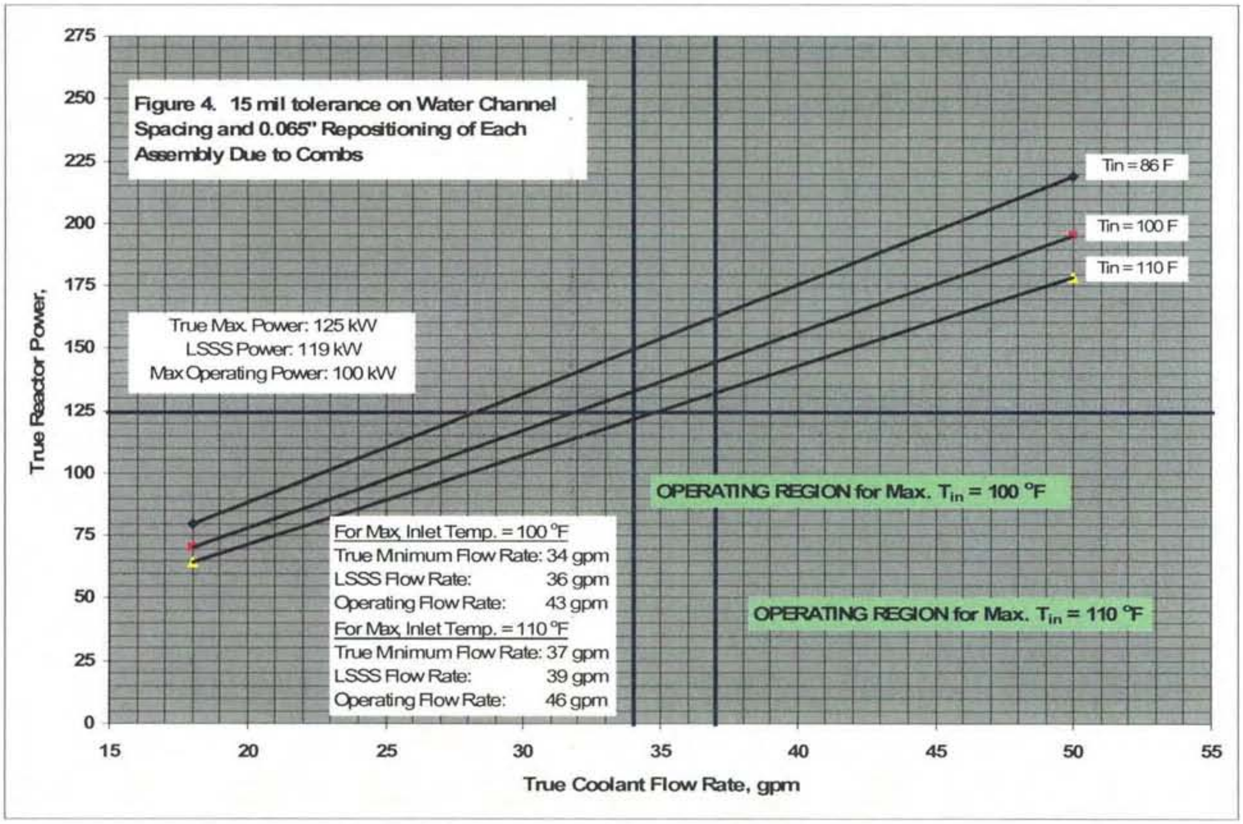


Figure 4-23 True Power versus True Flow Rate at Which ONB Occurs with 15 mil Tolerance on Water Channel Spacing and Repositioning of Assemblies in Fuel Box to Account for Thickness of Combs.

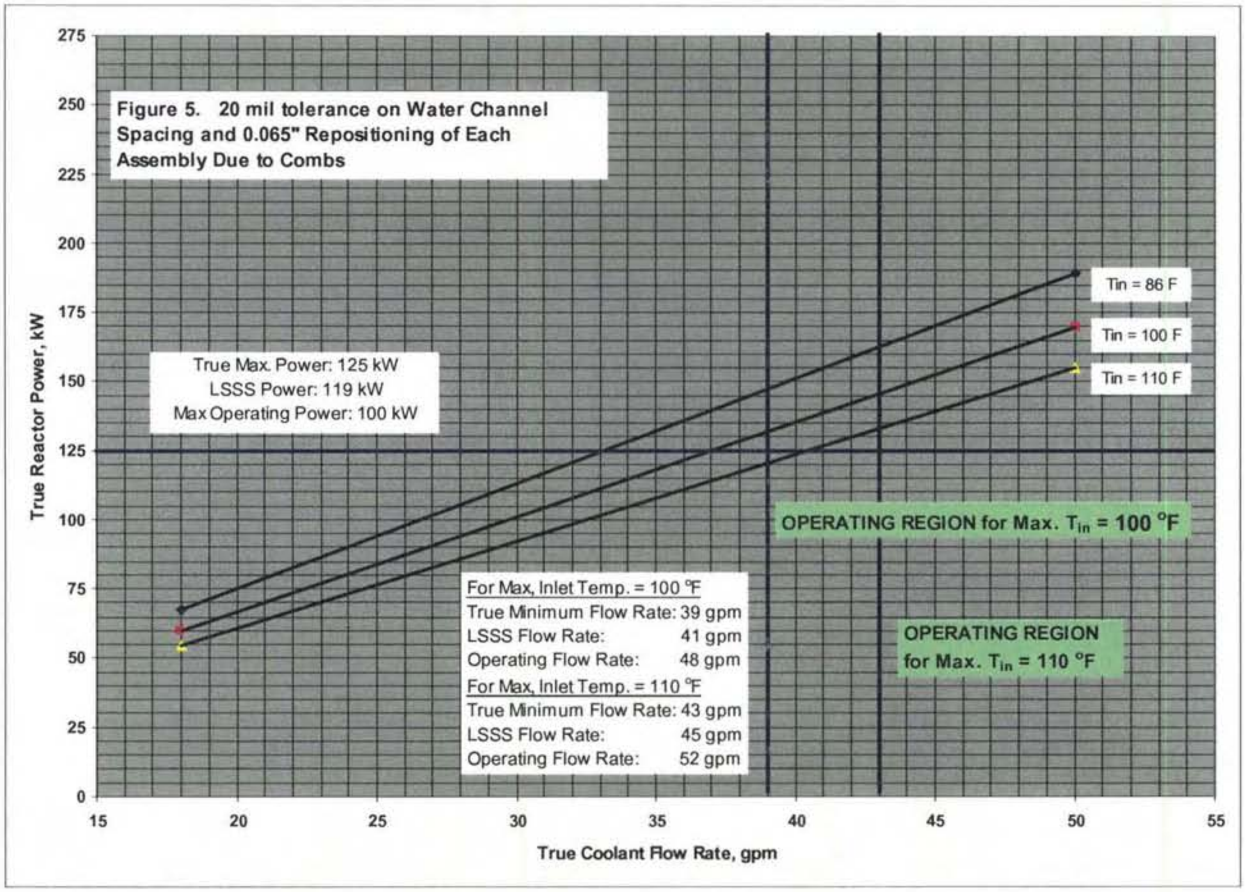


Figure 4-24 True Power versus True Flow Rate at Which ONB Occurs with 20 mil Tolerance on Water Channel Spacing and Repositioning of Assemblies in Fuel Box to Account for Thickness of Combs.

4.7.4 Proposed Parameters for LEU Core

Based on the results of Section 4.7.3.2, two sets of key flow rate settings are proposed, depending on the maximum value of the average inlet temperature:

1. **True maximum inlet temperature = 100 °F (LSSS on inlet temperature of 99 °F)**

Table 4-27 gives the proposed settings for different flow rates depending on the different water channel spacing tolerances.

Table 4-27 Flow Rate Settings for Different Water-Channel-Spacing Tolerances

Type of flow rate	Water Channel Spacing tolerance (mil)			
	1	10	15	20
True Min. (gpm)	34	34	34	39
LSSS (gpm)	36	36	36	41
Operating (gpm)	43	43	43	48

Note that only the case with 20 mil tolerance requires changes to the prior values.

2. **True maximum inlet temperature =110 °F (LSSS on inlet temperature of 109 °F)**

Table 4-28 gives the proposed settings for different flow rates depending on the different water channel spacing tolerances.

Table 4-28 Flow Rate Settings for Different Water-Channel-Spacing Tolerances

Type of flow rate	Water Channel Spacing tolerance (mil)			
	1	10	15	20
True Min. (gpm)	34	34	37	43
LSSS (gpm)	36	36	39	45
Operating (gpm)	43	43	46	52

Cases with 15 mil tolerance and with 20 mil tolerance require changes to the prior values. Since a measured minimum water channel spacing of 90 mils (20 mil tolerance) is anticipated on a number of the LEU fuel assemblies being re-worked by BWXT, the following values for key coolant flow rate parameters are proposed in Table 4-29. The UFTR Technical Specifications have been appropriately changed.

Table 4-29 Flow Rate Settings for 20 mil Water-Channel-Spacing Tolerance (measured minimum water channel spacing of 90 mils)

Type of flow rate	True Max. Inlet Temp.	
	100 °F	110 °F
True Min. (gpm)	39	43
LSSS (gpm)	41	45
Operating (gpm)	48	52

The accident analyses in Chapter 13 were performed using a true minimum coolant flow rate of 34 gpm, which is consistent with the proposed Technical Specifications for maximum water channel spacing tolerances of 10 mils and 15 mils. For a maximum tolerance of 20 mils, the accident analyses provides results that are more conservative than with the true minimum flow rate of 39 gpm proposed in the Technical Specifications.

It is recognized that the thermal-hydraulic analyses with the hot-channel factor method used in the analyses in this supplement give conservative results. ANL is currently evaluating alternative methods for reducing some of the conservatism due the somewhat-unique variations in the water channel spacing anticipated in the UFTR LEU fuel assemblies. Measurements of the minimum channel spacing in each LEU fuel assembly are anticipated before August 10, 2006. Using the measured minimum channel spacing and less conservative, but more accurate, factors to represent more realistic coolant flow rate through these channels, it may be possible to reduce the coolant flow rate settings shown in Table 5.

4.7.5 Effect of Handle on Margin to Onset of Nucleate Boiling

The UFTR HEU fuel design incorporates a handle that closes off the top of the central coolant channel. A similar handle was continued in the LEU fuel assembly design shown in Figure 4-25. The open sides of the coolant channels allow the flow to exit when the top of the channel is blocked by the handle. The questions are: How much is the flow in the central channel reduced by the presence of the handle? What effect does this reduced flow have on the margin to onset of nucleate boiling?

[redacted]

Figure 4-25 Top Section of an LEU Fuel Assembly

The PLTEMP calculations in Ref. 17 did not include the effect of the handle. To answer the above questions, a modification was made to the PLTEMP/ANL V3.0 code to allow any channel to have an additional K-loss factor. Explicit calculations using the PLTEMP code to determine the power level at which ONB occurs for added K-loss factors of 0 - 100 in the central channel only are shown in Figure 4-26. The reference point with an added K-loss of 0 from Figure 5 of Ref. 17 is a power level of 133 kW, a coolant flow rate of 39 gpm, and an inlet temperature of 100 F for a tolerance of 20 mils on channel spacing. Changes relative to this reference point will be very similar for other values of coolant flow rate and inlet temperature. For added K-loss values of 0 - 20 in the central channel, the first interior channel and not the channel with the handle is the limiting channel. The power at which ONB occurs increases slightly because reducing flow in the central channel slightly increases flow in the other channels. There is a transition region for additional K-loss values between 20 and 30, where the limiting channel switches to the central channel with the handle. For K values of 30 or more, the reduction of flow in the central channel results in a steadily decreasing power level at which ONB occurs.

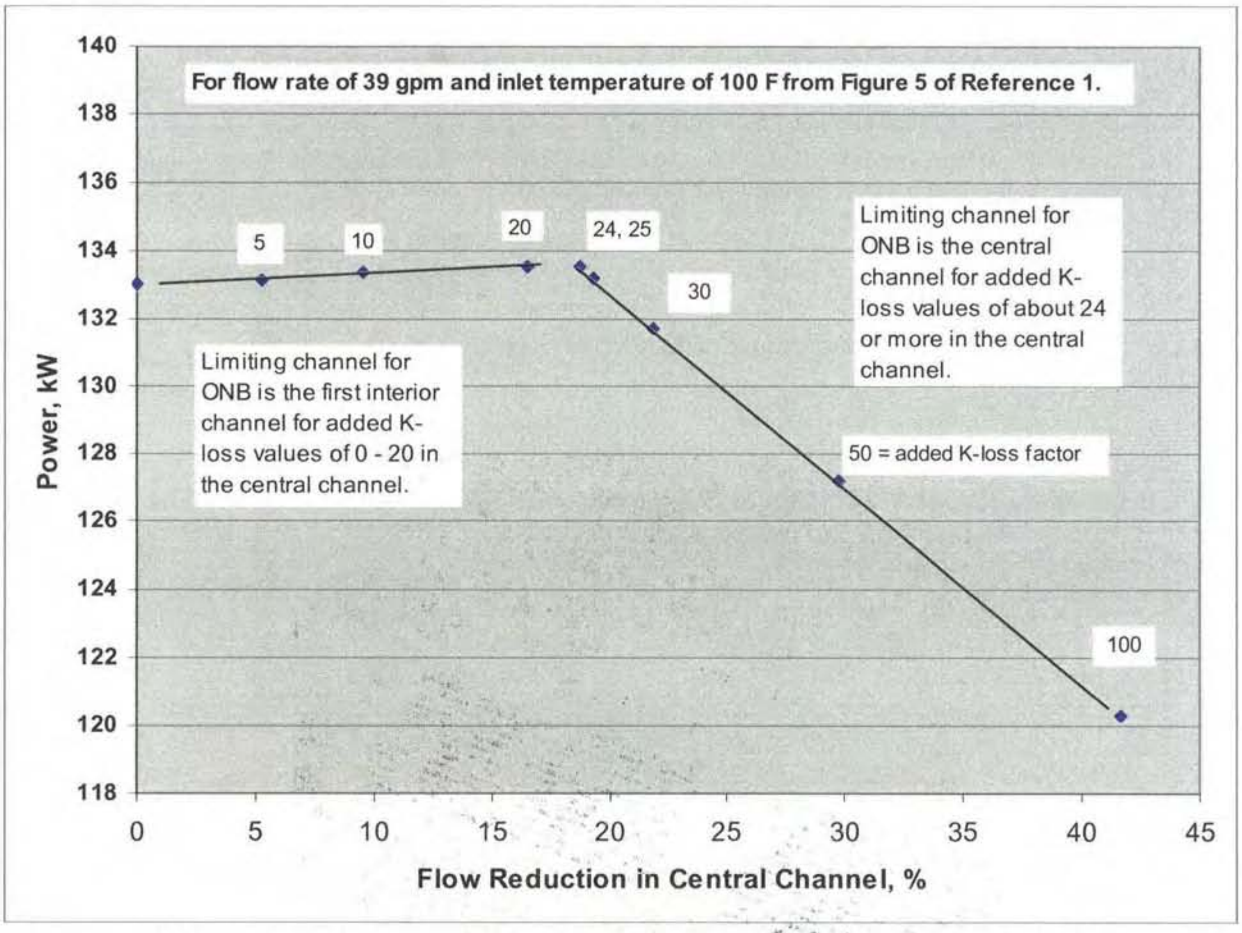


Figure 4-26 Power Level Versus Percent Reduction in the Flow Rate in Central Channel.

An estimate of the appropriate K-loss factor can be obtained by considering a duct consisting of a rectangular channel with a sharp corner in the turn (Ref. 18, p. 538). While these data do not match UFTR proportions exactly, they do provide a very good indication of the expected range of the data. The K-loss factor for this case in Ref. 18 is never more than 10, and is usually much less. Consequently, considering two 90 elbows,[3] it is conservative to assume that the added K-loss factor will be less than 20. Thus, it is expected that the power level at which ONB occurs remains the same or somewhat increases with the presence of the hand

[3] Representing two sharp turns for the flow in the central channel

5. Reactor coolant system

The HEU to LEU conversion does not require any changes to the reactor coolant system. More details about this topic can be found in Ref. 1.

6. Engineering Safety Features

The HEU to LEU conversion does not require any changes to engineering safety features. More details about this topic can be found in Ref. 1.

7. Instrumentation and control

The HEU to LEU conversion does not require any changes to instrumentation and control. More details about this topic can be found in Ref. 1.

8. Electrical power system

The HEU to LEU conversion does not require any changes to electrical power systems. More details about this topic can be found in Ref. 1.

9. Auxiliary system

Existing procedure will be used for fuel storage.

10. Experimental Facility and utilization

The HEU to LEU conversion does not require any changes to experimental facility and utilization of the UFTR. More details about this topic can be found in Ref. 1.

11. Radiation Protection and Radioactive Waste Management

The HEU to LEU conversion does not require any changes to the radiation protection and radioactive waste management of the UFTR facility. More details about this topic can be found in Ref. 1.

12. Conduct of Operation

12.1 Organization and Staff Qualification

The HEU to LEU conversion does not require any changes to the organization and staff qualification of UFTR personnel. More details about this topic can be found in Ref. 1.

12.2 Procedures

Few procedural changes are required for the HEU to LEU fuel conversion. The expected changes are outlined in the following paragraphs.

In SOP-A.7 (Determination of Control Blade Integral or Differential Reactivity Worth) the reference to 2.3% $\Delta k/k$ core excess reactivity limit will be changed in Section 3.1.1.1(2) to 1.4% $\Delta k/k$ per Technical Specification changes. The same change from 2.3% $\Delta k/k$ to 1.4% $\Delta k/k$ will be made for the total reactivity worth allowed for all experiments in Section 3.1.1.2(2). In addition, the power ratio curves in Appendix II of this procedure for evaluation of control blade reactivity worth by the blade drop method will need to be regenerated for the LEU fuel if this method of blade worth measurement is utilized.

In SOP-C.2 (Fuel Loading) the excess reactivity limit will be changed to 1.4% versus 2.3% in Section 3.2, Section 4.5, in the note for Section 7.2.2.5, in Section 7.4.1.1.3 and Section 7.4.2.1.3.

In SOP-C.3 (Fuel Inventory Procedure) the fuel inventory forms should be changed to account for the presence of LEU uranium-silicide fuel. Though this change is not required, it is recommended to assure ease of assuring the presence and location of all special nuclear material (fuel).

12.3 Operator Training and Re-qualification

The HEU to LEU conversion requires no changes to the requalification training program itself. Some changes will be required where fuel description, fuel loading, reactivity limitations and safety limits as well as accident analyses are addressed in the various training modules. These will be changed as the modules are updated for the respective training topics.

12.4 Emergency Plan

The only changes made for the approved UFTR Emergency Plan are the fuel description on page 1-1 in Section 1.3.1; in the Credible Accidents and Consequences in Section 1.5 on pages 1-6, 1-12 and 1-13; in the definition of Site Boundary on page 2.3; and possibly in Section 5.0 Emergency Action Levels on page 5-1 and Section 6.0 Emergency Planning Zone on page 6-1; though the latter two do not seem necessary at this point based on the analyses in Section B. The changes made have been submitted in a separate transmittal directly to the NRC.

12.5 Physical Security

The HEU to LEU conversion does not necessitate any changes at this point; changes are anticipated to be proposed but will be submitted under separate cover and withheld from public disclosure.

12.6 Reactor Reload and Startup Plan

Existing procedures will control unloading HEU and loading LEU fuel. The primary applicable procedures are SOP.C.1 (Irradiated Fuel Handling), SOP-C.2 (Fuel Loading), SOP-D.2 (Radiation Work Permit), and SOP-E.2 (Alterations to Reactor Shielding and Graphite Configuration). Of these, the only changes needed are that the core excess reactivity limit specified in SOP-C.2 will be changed from 2.3% Δk/k to 1.4% Δk/k. All other procedural steps remain unchanged as stated. A more detailed discussion is included in Appendix A.11.

13. Accident Analysis

The current UFTR Technical Specifications place limits on the core excess reactivity and the reactivity worth of experiments. Namely,

- The core reactivity at cold critical, without xenon poisoning, shall not exceed 2.3% Δk/k. [Section 3.1(2)]
- The absolute reactivity worth of any single moveable or nonsecured experiment shall not exceed 0.6% Δk/k. [Section 3.5(3a)]
- The total absolute reactivity worth of all experiments shall not exceed 2.3% Δk/k. [Section 3.5(3b)]

For the LEU core, UFTR proposes to alter these Technical Specifications to read:

- The core reactivity at cold critical, without xenon poisoning, shall not exceed 1.4% Δk/k. [Section 3.1(2)]
- The absolute reactivity worth of all moveable or nonsecured experiment shall not exceed 0.6% Δk/k. [Section 3.5(3a)]
- The total absolute reactivity worth of all experiments shall not exceed 1.4% Δk/k. [Section 3.5(3b)]

Rapid insertions of reactivity due to malfunction or failure of moveable or non-secured experiments are considered to be remote possibilities, but are analyzed here none-the-less. Scenarios in which all of the allowed excess reactivity, including that from secured experiments, is inserted rapidly into the core are considered to be incredible and are not analyzed here.

Based on the proposed changes to the Technical Specifications for the LEU core, two hypothetical reactivity insertion transients for the UFTR were postulated and analyzed:

- A rapid insertion of 0.6% Δk/k reactivity. This scenario represents the reactivity insertion resulting from the rapid ejection of the maximum worth of all moveable and nonsecured experiments from the reactor. Cases were analyzed both with and without reactor SCRAM.
- A reactivity ramp insertion of 0.06% Δk/k/second for 10 seconds. This scenario represents the insertion of reactivity due to control blade withdrawal at the maximum rate allowed by the UFTR Technical Specifications. This accident is assumed to be terminated by reactor SCRAM.

Calculations were performed for both the HEU and reference LEU cores. Reactivity coefficients and neutron kinetics parameters calculated as part of the reactor dynamic design (Section 4.5.5 were employed to account for inherent reactivity feedback mechanisms. Because of the non-linearity of the feedback coefficients with increasing temperature or water void, the calculated coefficients with the smallest magnitude were employed for conservatism in the transient analyses. The coefficients that were calculated are summarized in Table 13-1.

Table 13-1 Kinetics Parameters and Reactivity Coefficients Calculated for UFTR Accident Analyses.

Parameter	**HEU Core**	**Fresh LEU Core**	**Depleted LEU Core**
β_{eff}	0.0079	0.0077	0.0076
ℓ (μs)	187.4	177.5	195.1
C_{void} ($\Delta\rho$/%void)	-1.48E-03	-1.53E-03	-1.46E-03
C_{water} ($\Delta\rho/^oC$)	-5.91E-05	-5.68E-05	-5.26E-05
C_{fuel} ($\Delta\rho/^oC$)	-0.29E-05	-1.65E-05	-1.49E-05

The total reactivity worth of the safety and regulating blades calculated for the HEU and LEU cores was assumed to be inserted under reactor SCRAM conditions. The total blade reactivity worths calculated in Section 4.5 are for insertion from a fully-withdrawn position. However, it is difficult to envision a situation where all blades are fully-withdrawn when fuel is in the core. Rather, the reactivity worth of inserting all blades from a critical core condition was used for the SCRAM reactivity in this scenario. The worth of all blades when inserted from a critical core condition was calculated to be 5.44% Δk/k for the HEU core and 5.09% Δk/k for the LEU core.

The UFTR places trip settings (Limiting Safety System Settings) on several measurable parameters to ensure that the safety limits are not violated.

- Trip settings are placed on reactor power. There are two power measurement channels. In the current Technical Specifications, the reactor trips if the power reaches 125 kW. For the LEU core, it is proposed to change this trip setting to a power of 119 kW, taking into account a 5% uncertainty in the measurement of the reactor power.
- The UFTR is tripped when the reactor period becomes less than 3 seconds. There is a single channel for measuring the reactor period.
- Trip settings are placed on the coolant flow rate. In the current Technical Specifications, the reactor trips if the coolant flow rate becomes less than 30 gallons/minute. It is proposed to increase this trip setting to 36 gallons/minute for the LEU core, including a 5% uncertainty on measurements of coolant flow rate. The true minimum value of the coolant flow rate is 34 gpm.
- In the current Technical Specifications, a trip setting is placed on the average coolant temperature at the outlet of any fuel box; this setting is 155 °F (68.3 °C). For the LEU core, UFTR proposes to place a trip setting on the average primary coolant inlet temperature, as well as the trip setting on the coolant outlet temperature. The trip setting on the average coolant inlet temperature is proposed to be 109 °F (42.8 °C), taking into account an uncertainty of 1 °F in the measurement of this parameter. The true value of the maximum average primary inlet temperature is 110 °F. The true maximum power level is 125 kW.

The trip settings which are considered applicable to each of the accident analyses are discussed in each section below. When measurements indicate that the reactor has reached one of the trip settings, a signal is sent to release the blades and drop them into a fully-inserted position. A delay time of 100 ms (0.1 seconds) was used to represent the delay from the time the SCRAM signal is sent to when the blades actually start to fall. The blade drop time is required to be less than 1 second in the current UFTR Technical Specifications. Blade drop times of both 1.0 and 1.5 seconds were considered in these analyses, in anticipation of increasing the Technical Specification for the blade drop time to 1.5 seconds.

13.1 Reactivity Insertion Accidents

13.1.1 Rapid Insertion of 0.6% Δk/k

This hypothetical accident was analyzed using the RELAP5-3D code (Ref. 10). The reactor was modeled as two fuel-plate channels. One channel represented the fuel plate and associated coolant for the plate with peak power in the core, and the other channel represented the average of the remainder of the core. The fuel power density profiles for the peak and average channels were taken from MCNP results for the core operating at 100 kW. Fuel plate and channel dimensions corresponding to the fuel assembly design were utilized. The effect of engineering uncertainties (hot channel factors) was not included in these analyses. A comparison of key parameters for the HEU and LEU cores is provided in Table 13-2.

Table 13-2 Selected Parameters for UFTR HEU and LEU Core Transient Analyses.

Parameter	HEU Core	LEU Core
Fuel meat thickness, mm	[redacted]	[redacted]
Fuel plate thickness, mm	[redacted]	[redacted]
Fuel plate width, mm	[redacted]	[redacted]
Fuel plate heated length, mm	[redacted]	[redacted]
Coolant channel thickness, mm	3.48	2.82
Number of axial nodes in fuel plate	12	12
Fraction of core represented by peak power channel	0.42%	0.31%
Coolant inlet temperature, °C	30	30, 30, 42.8
Coolant exit pressure, kPa	101.3	101.3
Coolant inlet flow rate, gpm	43	43, 34, 34
Initial total core power, kW	100	100
Peak-to-average power density	1.62	1.78

Three different steady-state thermal-hydraulic conditions were considered to evaluate the range on the UFTR accident response under different initiating conditions. First, the reactivity insertion accident was analyzed for the nominal coolant inlet temperature of 30°C (86°F) and the nominal flow rate of 43 gallons/minute. In addition to evaluating the accident response at nominal conditions, the impact of the reactivity insertion when the UFTR coolant flow rate is at the proposed low-flow trip limit of 34 gallons/minute was considered. Lastly, the accidents were analyzed with the coolant at the low-flow trip setting and the inlet temperature at the proposed trip setting of 109 °F (42.8 °C).

When the reactivity is inserted rapidly as in this case, the reactor period immediately drops below the trip setting of 3 seconds. However, assuming a single failure to add conservatism to the accident analysis, the period trip is assumed to fail. The next trip reached in the sequence is the 125 kW limit on total reactor power. After an assumed 100 ms delay, the control blades begin to drop into the core.

Results for the rapid reactivity insertion accidents with SCRAM are summarized in Table 13-3. The HEU core reaches a maximum power of about 290 kW and the peak clad temperature increases to no more than 54°C. Based on the RELAP5 analysis of the steady state, this transient results in a fuel temperature rise of only 1°C. If the control blade drop time is increased from 1.0 second (the current Technical Specification for the UFTR) to 1.5 seconds, there is practically no impact on the maximum fuel and clad temperatures (<0.1°C).

For the protected rapid reactivity insertion accident in the fresh LEU core, the core power increases to 316 kW before the transient is terminated by insertion of the control blades as a result of the overpower trip. For the depleted core, the peak power is slightly higher, 322 kW. The coolant flow rate and inlet temperature do not affect the peak power of the core in the accident when the reactor control system is operational. The peak local clad temperature is naturally higher for those cases with lower flow and higher inlet temperature, but it is no higher than 68°C in all cases. The temperature increase due to the reactivity insertion in the LEU core is about 1°C. If the blade drop time increases from 1.0 to 1.5 seconds, there is practically no impact on the temperature increase (<0.3°C).

Table 13-3 RELAP5-3D Results for Rapid Insertion of 0.6% Δk/k in UFTR with SCRAM[1].

Core	**HEU**		**Fresh LEU**				**Depleted LEU**			
P_o (kW)	100	100	100	100	100	100	100	100	100	100
Steady State Condition	43 gpm, T_{in}=86°F	43 gpm, T_{in}=86°F	43 gpm, T_{in}=86°F	34 gpm, T_{in}=86°F	34 gpm, T_{in}=109°F	43 gpm, T_{in}=86° F	43 gpm, T_{in}=86° F	34 gpm, T_{in}=86° F	34 gpm, T_{in}=109°F	43 gpm, T_{in}=86°F
Blade Drop Time (s)	1.0	1.5	1.0	1.0	1.0	1.5	1.0	1.0	1.0	1.5
Time to Peak Power (s)	0.17	0.17	0.14	0.14	0.14	0.14	0.14	0.14	0.14	0.15
Peak Power (kW)	291	295	316	316	316	318	322	322	322	328
$T_{fuel,max}$ at Peak Power (°C)	54.1	54.1	51.9	54.4	66.7	51.9	52.0	54.8	67.0	52.1
$T_{fuel,max}$ (°C)	54.4	54.6	52.2	54.8	67.0	52.5	52.6	55.3	67.5	52.6
$T_{clad,max}$ (°C)	54.4	54.5	52.2	54.7	67.0	52.5	52.6	55.3	67.5	52.5
$T_{cool,max}$ (°C)	44.4	44.4	44.6	47.6	59.9	44.6	44.5	47.5	59.8	44.5

[1]Assumes single failure of reactor control system. For the rapid reactivity insertion accidents, the reactor period trip was assumed to fail.

Cases without reactor SCRAM were also evaluated for the HEU and LEU cores. Results for the RELAP5-3D analyses of an unprotected insertion of 0.6% Δk/k in 100 ms on the HEU and LEU cores are presented in Table 13-4. The HEU core power was calculated to quickly rise from a steady-state power of 100 kW to 1302 kW before inherent reactivity feedback mechanisms (predominantly from coolant voiding) suppress the transient power spike. The local peak clad temperature in the HEU core reaches 89°C at the time of peak power, and continues to rise to a maximum of 108°C.

Table 13-4 RELAP5-3D Results for Rapid Insertion of 0.6% Δk/k in UFTR without SCRAM.

Core	HEU	Fresh LEU			Depleted LEU		
P_o (kW)	100	100	100	100	100	100	100
Steady State Condition	43 gpm, T_{in}=86°F	43 gpm, T_{in}=86°F	34 gpm, T_{in}=86°F	34 gpm, T_{in}=109°F	43 gpm, T_{in}=86°F	34 gpm, T_{in}=86°F	34 gpm, T_{in}=109°F
Time to Peak Power (s)	2.57	2.48	2.44	2.30	2.36	2.32	2.19
Peak Power (kW)	1302	1199	1186	1112	1337	1321	1235
$T_{fuel,max}$ at Peak Power (°C)	89	95	95	100	96	96	101
$T_{fuel,max}$ (°C)	108	107	108	109	108	109	110
$T_{clad,max}$ (°C)	108	107	108	109	108	109	110
$T_{cool,max}$ (°C)	101	101	101	102	101	101	102

For the LEU core, the response to this accident for both fresh and depleted cores was evaluated. The largest power increase occurs under the nominal thermal-hydraulic conditions, reaching 1199 and 1337 kW for the fresh and depleted cores, respectively. In both cases, the local clad temperature reaches a maximum of about 108°C. Under the limiting conditions of low flow and high inlet temperature, the transient terminates sooner because coolant voiding, which is a strong feedback mechanism, occurs sooner when the steady-state temperatures are higher; the maximum clad temperature reached under these conditions is 2°C higher. In all cases, the maximum clad temperature is well-below the Safety Limit of 530°C (986°F) for Al-6061 clad. Thus, even without action of the reactor control system, the UFTR can tolerate the sudden ejection of all moveable and non-secured experiments (limited to 0.6% Δk/k by the proposed Technical Specifications) without any fuel damage.

The unprotected insertion of 0.6% Δk/k was modeled for 300 seconds to show that the power does not rise again after the suppression of the initial power spike. Instead, the core power declines to an equilibrium power level of about 600 kW after the spike. Under these conditions, the coolant reaches the saturation temperature and boiling occurs in the uppermost nodes of the coolant channel. However, the peak temperatures of about 108°C in the fuel and cladding for the LEU core are well below the Safety Limit of 530°C.

13.1.2 Slow Insertion of 0.06% Δk/k/second

The UFTR Technical Specifications require that the reactivity addition from control blade withdrawal must be less than 0.06% Δk/k/second when averaged over a 10 second interval. In this hypothetical accident, a reactivity insertion at this maximum rate initiates the transient, and continues until the reactor is tripped by the control system. Because of the slower reactivity insertion rate in this accident compared with the rapid insertion accident, the reactor period remains longer than the 3 second trip setting throughout the transient. Because there are two power measurement channels, however, the overpower trip of 125 kW can SCRAM the reactor, even assuming failure of one of the power trips. The results of RELAP5-3D calculations in the HEU and LEU cores are summarized in Table 13-5.

The HEU core reaches the overpower trip setting of 125 kW within about 2.1 seconds, and the total core power increases to 127 kW before the transient is terminated by the control system. The maximum temperatures in the fuel and cladding of the HEU core increase by less than 1°C. The results for the fresh and depleted LEU cores are similar: the core power increases to 127 kW during the accident, and the clad temperature increases by only a fraction of a degree from the steady-state condition.

Figure 13-1 shows the core power and peak local clad temperature for this hypothetical accident scenario for the HEU core. The core power trace shows that there is almost no impact of a 1.5 second blade drop time versus a drop time of 1.0 second specified in the current Technical Specifications. The effect on the maximum fuel and clad temperatures in the HEU core is negligible.

Figure 13-2 shows the peak local clad temperature in the depleted LEU core for this accident scenario. The clad temperature is naturally higher for the lower flow and higher inlet temperature conditions. The clad temperature rise due to the reactivity insertion is small. The consequence of a 1.5 second blade drop time vs. 1.0 second is negligible.

Table 13-5 RELAP5-3D Results for Slow Insertion of 0.06% Δk/k/second with SCRAM[1].

Core	HEU		Fresh LEU				Depleted LEU			
P_o (kW)	100	100	100	100	100	100	100	100	100	100
Steady State Condition	43 gpm, T_{in}=86°F	43 gpm, T_{in}=86°F	43 gpm, T_{in}=86°F	34 gpm, T_{in}=86°F	34 gpm, T_{in}=109°F	43 gpm, T_{in}=86°F	43 gpm, T_{in}=86°F	34 gpm, T_{in}=86°F	34 gpm, T_{in}=109°F	43 gpm, T_{in}=86°F
Blade Drop Time (s)	1.0	1.5	1.0	1.0	1.0	1.5	1.0	1.0	1.0	1.5
Time to Peak Power (s)	2.26	2.26	2.22	2.22	2.22	2.22	2.18	2.18	2.18	2.18
Peak Power (kW)	127	127	127	127	127	127	127	127	127	127
$T_{fuel,max}$ at Peak Power (°C)	54.2	54.2	52.1	54.6	66.8	52.1	52.1	54.9	67.1	52.1
$T_{fuel,max}$ (°C)	54.2	54.2	52.1	54.6	66.8	52.1	52.1	54.9	67.1	52.1
$T_{clad,max}$ (°C)	54.2	54.2	52.0	54.6	66.8	52.0	52.1	54.9	67.1	52.1
$T_{cool,max}$ (°C)	44.5	44.5	44.6	47.6	60.0	44.6	44.5	47.5	59.9	44.5

[1]Assumes single failure of reactor control system. For the slow reactivity insertion accidents, one of the reactor power trip channels was assumed to fail, but the second channel functioned correctly.

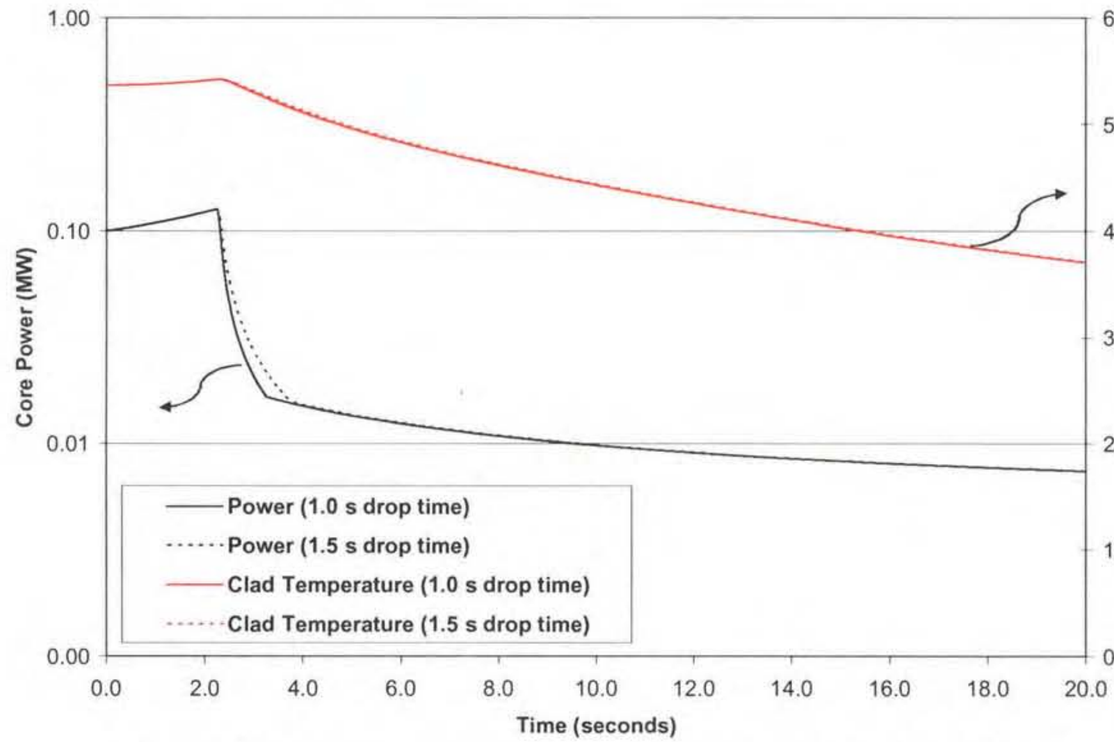


Figure 13-1 Slow Insertion of 0.06% Δk/k/s with SCRAM in HEU Core

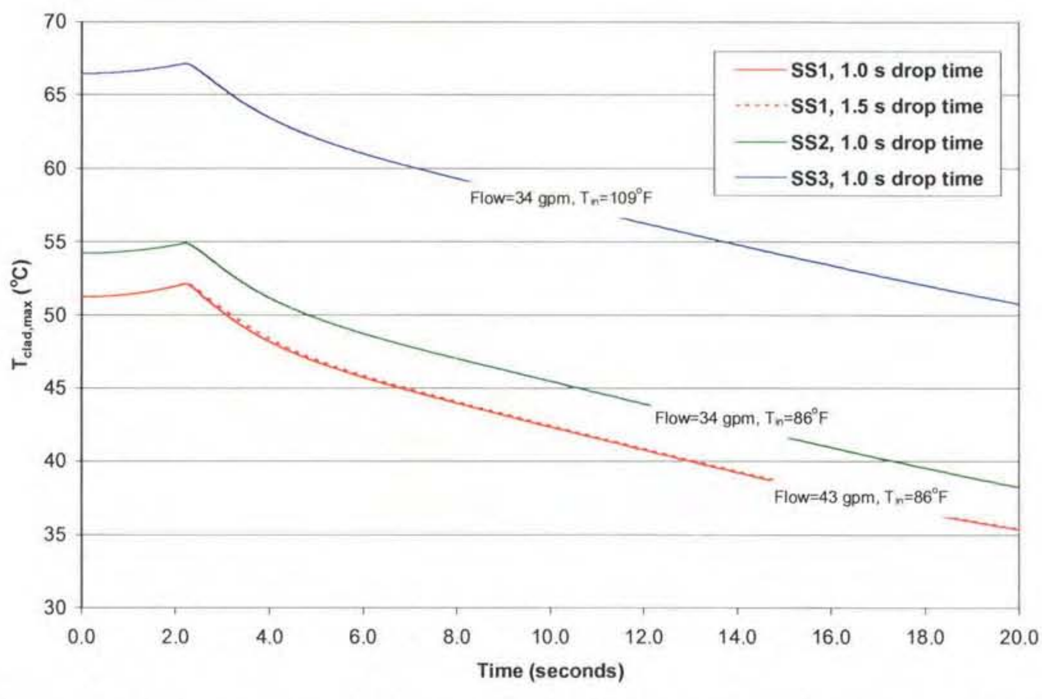


Figure 13-2 Slow Insertion of 0.06% Δk/k/s with SCRAM in Depleted LEU Core

13.2 Loss-of-Coolant Accident

The UFTR FSAR (Ref. 1) evaluated a loss-of-coolant accident during full-power operation. The increase in fuel temperature following a loss-of-coolant and shutdown of the reactor either by the negative void coefficient of reactivity or by the insertion of control blades into the reactor showed that the fuel temperature will increase by less than 17°C (30°F) following a full trip event (blade drop with coolant dump).

This loss-of-coolant accident was not re-evaluated here for the LEU core because the average power per fuel plate in the fuel element with highest power in the LEU core (Table 4-11, position 2-3) is only 75% of the average power per plate in the corresponding fuel element (position 5-1) in the HEU core. This is mostly because an LEU element contains 14 fuel plates and an HEU fuel element contains 11 fuel plates. In addition, the volume fraction of air available for cooling after the water is lost is slightly larger in an LEU fuel element than in an HEU fuel element. These two effects, the lower power per plate and the slightly larger coolant volume fraction, will result in a fuel temperature increase in the LEU core that is less than the approximate 17°C temperature increase in the HEU core.

Consequently, a loss-of-coolant accident from operation at 100 kW power in the LEU core will result in maximum fuel and clad temperatures that are far below the safety limit of 530 °C. Integrity of the cladding will be maintained and there will be no release of radioactivity.

13.3 Fuel Handling Accident (FHA)

This hypothetical accident assumes that one fuel element is dropped during a core reload or other fuel handling operation. Fuel handling operations allow moving only one bundle at a time and it must be secured before proceeding to move another. For this event, assumptions were made based on credible operation of the reactor. Typically, the UFTR is shutdown from power operation for more than seven days prior to commencing fuel-handling operations. In all cases, the reactor would be shutdown from power operation for at least three days to allow substantial decay of fission product inventory. Since the coolant water may be drained from the core immediately after shutdown, any fission product release would be directly to the air of the reactor cell as a conservative measure. Tech Spec 4.2.7 is augmented to require at least three days to pass after UFTR power operations before not only fuel handling but also before moving the last two layers of concrete blocks to access the fuel. This new Tech Spec would limit the possible/potential consequences of fuel handling accidents and preclude damaging a fuel bundle with a dropped shield block before three days have elapsed.

The following data and assumptions were used to evaluate the source term associated with this accident:

(1) The reactor is operated at 100 kW steady-state power for 4 hours per day for 30 days.

(2) The fuel elements with highest power in the HEU and LEU cores based on MCNP5 calculations were selected for evaluation. These were bundle 5-1with a power of 5.77 kW in the HEU core and bundle 2-3 with a power of 5.45 kW in the LEU core. These bundle powers were derived from an MCNP tally that assumes that all of the energy produced is deposited locally. This assumption results in maximum bundle powers that are slightly larger

than those shown in Table 4-14, which accounts for energy deposited in the fuel plates and gamma energy deposited in the coolant, moderator, and structural materials.

(3) Radioisotope inventories were calculated three days after shutdown from power operation. To ensure that the three day period is satisfied, UFTR has augmented Technical Specification 4.2.7 (1) for the LEU core to require at least three days to pass after UFTR power operations before fuel handling. The HEU core was analyzed with the same assumption for comparison purposes.

(4) The radioisotopes of greatest significance for release in case of an accident are the radioiodines and the noble gases, krypton and xenon.

(5) It is postulated that the fuel element would undergo severe mechanical damage due to being dropped during the fuel handling operation and that this damage would be sufficient to expose fuel surface areas equivalent to stripping the aluminum cladding from one fuel plate out of 11 plates in an HEU element and one fuel plate out of 14 fuel plates in an LEU element. It is further assumed (Ref. 19) that 100% of the gaseous activity produced within the recoil range of the particles (1.37 x 10^{-3} cm) or 2.7% of the total volatile activity instantaneously escapes from the fuel plate into the reactor cell.

13.3.1 Radionuclide Inventories

Radionuclide inventories for the highest power fuel element in the HEU and LEU cores core were calculated using the ORIGEN-S code (Ref. 6) under the assumptions in the previous paragraph. Additional calculations verified that all of the gaseous fission products except ^{85}Kr (half-life 11 years) reach their equilibrium concentrations based on this operating assumption.

The activities of the krypton, iodine, and xenon isotopes in one fuel plate of the HEU and LEU cores are given in Table 13-6 along with the inventory (2.7% of the total) that is assumed to escape from the damaged fuel plate into the air of the reactor cell.

Table 13-6 Calculated Radionuclide Inventories (Ci) Released into the Reactor Cell from the FHA in the HEU and LEU Cores

	HEU Core		LEU Core	
Isotope	**Ci in One Plate Three Days after Shutdown**	**Ci in 2.7 % of One Plate Three Days after Shutdown**	**Ci in One Plate Three Days after Shutdown**	**Ci in 2.7 % of One Plate Three Days after Shutdown**
Kr 85	1.16E-03	3.13E-05	8.58E-04	2.32E-05
Kr 85m	3.99E-05	1.08E-06	2.95E-05	7.95E-07
Kr 88	2.28E-07	6.17E-09	1.69E-07	4.56E-09
I 130	4.50E-06	1.21E-07	3.31E-06	8.95E-08
I 131	1.61	4.35E-02	1.19	3.22E-02
I 132	1.85	5.00E-02	1.37	3.70E-02
I 133	0.62	1.67E-02	0.46	1.23E-02
I 135	5.22E-03	1.41E-04	3.86E-03	1.04E-04
Xe 133	3.96	1.07E-01	2.92	7.90E-02
Xe133m	0.092	2.47E-03	0.068	1.83E-03
Xe 135	0.12	3.20E-03	0.088	2.37E-03
Xe135m	8.52E-04	2.30E-05	6.30E-04	1.70E-05

13.3.2 Methodology for Dose Calculations

The following assumptions and methods were used to calculate the doses for occupational exposure at the reactor site and public exposure near the reactor site. Doses were calculated for the most exposed member of the public in the unrestricted environment and the integrated exposure at the nearest permanent residence. As stated in NUREG-1537 Part 2 (Ref. 16), the accidental dose limits found acceptable to the NRC staff for reactors initially licensed before January 1, 1994, has been 5 rem to the whole body and 30 rem to the thyroid for occupational exposure and 0.5 rem to the whole body and 3 rem to the thyroid for members of the public.

<u>Occupational Exposure</u>

NUREG-1537, Part 1, Section 13.2 (7) (Ref. 20) provides guidance on the location for occupational exposure. Exposure conditions should account for the facility staff, including staff evacuation and reentry, until the situation is stabilized. According to this guidance, the location is inside the reactor building. This represents the immediate surrounding of the UFTR reactor which could be rapidly evacuated and controlled. The distance from the accident source would be 1.5 m to 9 m (5 ft to 30 ft) which is the approximate distances inside the reactor building.

The calculational methodology for the occupational assessment is based on the analysis presented in NUREG/CR-2079 (Ref. 19). The following assumptions were used in this analysis:

1. Breathing rate: 3.33×10^{-4} m^3/sec.
2. χ / Q, atmospheric dispersion factor: 0.01 sec/m^3 for the short distance from 1.5 to 9 m.
3. Inhalation activity fraction to the thyroid: 0.23
4. Fractional release from the fuel plate inventory scenario: 2.7%

5. Dose coefficients for iodine taken from Federal Guidance Report (FGR) No. 11 (Ref. 21) based on inhalation.
6. The radionuclide inventory release is assumed to occur over a 1 hour period.

The analysis in NUREG/CR-2079 uses 0.23 of the fraction of activity inhaled from the iodine that reaches the thyroid. The fraction value is obtained from an ICRP (1960) publication referenced in NUREG/CR-2079. The dose results are given as a dose rate, which can be used to assess the evacuation and reentry of facility staff.

Public Exposure

NUREG-1537, Part 1, Section 13.2 (7) (Ref. 20) provides guidance on the location for public exposure. Doses were calculated for the most exposed member of the public in the unrestricted environment and the integrated exposure at the nearest permanent residence. Following the release of radioactive material, a worker must evacuate the reactor cell through the access door on the west side of the reactor building creating a potential ground release pathway to the west fenced area. The distance from the access door to the fence is 16.5 m. The East Hall Housing facility is the nearest permanent residence, located 190 m from the outside wall of the reactor building.

Public Exposure - Maximum

The calculation methodology for assessment of the most exposed member of the public is based on the analysis from Ref. 20. The location of the most exposed member of the public is at the west fenced area, 16.5 m the reactor building. The following assumptions were used in this analysis:

1. Breathing rate: 3.47×10^{-4} m^3/sec for 0 to 8 hours based on Ref. 21
2. χ / Q, atmospheric dispersion factor: 0.01 sec/m^3 for the short distance of 16.5 m at the fence, similar to the occupational exposure.
3. Inhalation activity fraction to the thyroid: 0.23
4. Fractional release of iodine from the building: 0.25 based on Ref. 21
5. Fractional release from the fuel plate inventory scenario: 2.7%
6. Reactor cell leak rate: 10% to 20% volume / hour.
7. Dose coefficients for iodine taken from Federal Guidance Report (FGR) No. 11 (Ref. 20) based on inhalation.
8. Dose coefficients for xenon and krypton taken from FGR No. 12 (Ref. 22) based on dose coefficients for air submission.
9. Dose calculations are calculated for a 2 hour exposure. The exposure time is based on a reasonable amount of time it would take to have the member of the public evacuated from the area near the reactor building.

The public exposures will occur outside the reactor building which results in a fractional release of the iodines. Regulatory Guide 1.4 (Ref. 23) provides guidance on the amount of Iodine which will escape from the reactor building.

A reactor cell leak rate measured by UFTR is 11.5% volume/hr when the fan system is operating in the reactor building. A parametric study of the effect of the leak rate on the calculated doses was investigated. Leak rate values of 10%, 20% and 100% volume/hr were chosen. The effect of the varying leak rate is presented in Tables 13-10 and 13-15 for the FHA and MHA accidents, respectively. At 100% volume/hr the doses are approximately 98% of their maximum values. Thus 100% volume/hr leak rate essentially represents the maximum possible dose with a very large leak rate. The actual leak rate would be in the range of 10% to 20% volume/hr or possibly less when the fan system is not operating. The dose values for the lower leak rates would be representative of the maximum doses that would be expected to occur.

Public Exposure – Nearest Residence

The calculation methodology used for the nearest public permanent residence assessment is based on the analysis from Ref. 20. The location of the nearest permanent residence is at a distance of 190 m. The assumptions for the analysis are based on the following:

1. Breathing rate: 3.47×10^{-4} m^3/sec for 0 to 8 hours based on Ref. 21
2. Breathing rate: 1.75×10^{-4} m^3/sec for 8 to 24 hours based on Ref. 21
3. χ / Q, atmospheric dispersion factor is based on a ground release and local weather conditions. Values used in the analyses are given in Table 13-8 for a distance of 190 m.
4. Fractional release from the fuel accident inventory scenario: 2.7%.
5. Fractional release of iodine from the building: 0.25 based on Ref. 21
6. Reactor cell leak rate: 10% to 20% volume / hour.
7. Dose coefficients for Iodine taken from Federal Guidance Report (FGR) No. 11 (Ref. 20) based on inhalation.
8. Dose coefficients for xenon and krypton taken from FGR No. 12 (Ref. 22) based on dose coefficients for air submission.
9. Dose calculations are calculated for a 1 day (24 hours) exposure. The exposure time is based on a reasonable amount of time it would take to have the public evacuated from the nearest permanent residence.

The atmospheric dispersion factor (χ / Q) was calculated based on local weather and guidance provided in USNRC Regulatory Guide 1.4 (Ref. 21). The weather data for the University of Florida was collected by the Department of Physics at the University of Florida Main Campus. A synopsis of the weather data from July 2004 to July 2005 is given in Table 13-7 (Ref. 24). The table provides the wind speed and Pasquill condition and other weather data at the campus for monthly, quarterly and yearly average based on recent weather conditions.

Regulatory Guide 1.4 has atmospheric diffusion equations in section g(1) and g(2) for time periods for 0-8 hours and greater than 8 hours, respectively. Both diffusion equations in section g(1) and g(2) of Ref. 21 are based on a ground level release. The exposure analysis for the public at the nearest permanent residence uses a total of 24 hours of exposure. The resulting atmospheric dispersion factors for a distance of 190 m with a ground release were calculated based on the local weather. The most conservative weather condition from Table 13-7 is the monthly average from May 2005 with a wind speed of 1.58 m/s and a Pasquill Type B condition. These conditions will result in the highest atmospheric dispersion factors.

Table 13-7 University of Florida Main Campus Weather Data from July 2004 through July 2005

Monthly, Quarterly, & Yearly Atmospheric Averages for 24 hours: July 04-July05						
Months & Quarters	**Temp**		**Wind Direction**	**Wind Speed**		**Pasquill Cond.**
	F	**C**	**Degrees**	**mph**	**m/s**	
July-04	82.09	27.83	179.50	3.22	1.44	A
August-04	80.94	27.19	170.00	3.32	1.49	A
September-04	79.50	26.39	128.26	6.04	2.70	B
October-04	74.80	23.78	146.39	6.97	3.12	B
November-04	67.44	19.69	134.46	3.52	1.57	A
December-04	56.81	13.78	137.10	3.64	1.63	A
January-05	59.59	15.33	144.72	3.70	1.66	A
February-05	61.03	16.13	173.63	4.49	2.01	B
March-05	62.19	16.77	229.55	4.99	2.23	B
April-05	66.42	19.12	191.18	4.39	1.96	B
May-05	73.63	23.13	176.44	3.54	1.58	B
June-05	78.93	26.07	158.98	3.55	1.59	A
July-05	81.70	27.61	181.43	3.63	1.62	A
Quarterly Jul04-Sep04	80.86	27.14	159.57	4.18	1.87	A
Quarterly Oct04-Dec04	66.34	19.08	139.37	4.72	2.11	B
Quarterly Jan05-Mar05	60.79	16.00	182.93	4.39	1.96	A
Quarterly (+1 month) April05-Jul05	74.08	23.38	176.27	3.80	1.70	A
Yearly (+1 month) Jul04-Jul05	70.52	21.40	164.54	4.27	1.91	A

The values for the atmospheric dispersion factor (χ / Q) for the local site weather are given in Table 13-8. Values are calculated for distances of 190 m and 100 m. The 190m distance is the location of the nearest permanent residence and 100 m is the smallest distance the atmospheric dispersion factor can be determined. The values of σ_y and σ_z, which are the horizontal and vertical standard deviation of the plume, respectively, are found in Ref. 25, as indicated in Regulatory Guide 1.4. The reason for the limit on the minimum distance of 100 m is due to the data provided in Ref. 25 for σ_y and σ_z plume deviations. For distances less than 100 m, the atmospheric dispersion factor is conservatively assumed to be 0.01 sec/m^3 as given in NUREG/CR-2079 (Ref.19).

Table 13-8 Atmospheric Dispersion Factor (χ / Q) based on University of Florida Main Campus Weather

Distance (m)	σ_y (m)	σ_z (m)	χ / Q for 0 to 8 hours (sec / m^3)	χ / Q for 8 to 24 hours (sec / m^3)
190 (East Hall)	30.7	19.2	0.000341	0.000352
100 (Minimum dist.)	16.5	10.7	0.00114	0.00120

The dose conversion factors that were used to calculate thyroid and whole body doses for the occupational and public exposures are shown in Table 13-9. The values are obtained from Ref. 21 and Ref. 22. The half life values are given for each isotope and are used to determine the effect of the reactor cell leak rate. Ref. 26 discusses the methodology for calculating the effect of the leak rate on the dose.

Table 13-9 Half Life and Dose Conversion Factors Utilized for Analyses

Isotope	Half Life (sec)	Exposure-to-Dose Conversion Factors[20] for Inhalation of the Thyroid[a,b] (rem/Ci)	Effective Absorbed Energy per Disintegration[17] for the Whole Body[a] (MeV)	Dose Coefficients for Air Submersion[21] for the Effective Whole Body[b] (rem-s/Ci-m³)
Kr 85	3.40E+08		1.20E-04	4.40E-04
Kr 85m	1.61E+04		1.60E-04	2.77E-02
Kr 88	1.02E+04		6.84E-04	3.77E-01
I 130	4.45E+04	7.36E+04		
I 131	6.93E+05	1.08E+06		
I 132	8.26E+03	6.44E+03		
I 133	7.49E+04	1.80E+05		
I 135	2.37E+04	3.13E+04		
Xe 133	4.53E+05		6.44E-05	5.77E-03
Xe 133m	1.89E+05		1.01E-04	5.07E-03
Xe 135	3.29E+04		2.41E-04	4.40E-02
Xe 135m	9.14E+02		1.55E-04	7.55E-02

[a] Occupational exposure
[b] Public exposure

13.3.3 Dose Calculation Results for Fuel Handling Accident

The calculated thyroid doses and whole body doses for the occupational and public exposures for the fuel handling accident are shown in Table 13-10.

Table 13-10 Summary of Dose Results for the FHA in the HEU and LEU Cores

Occupational Radiological Exposure Rate from the HEU Core

Distance	Thyroid Dose		Whole Body Dose	
	Rate (rem / hr)	5 Minute Exposure (rem)	Rate (rem / hr)	5 Minute Exposure (rem)
Inside Reactor Building	0.0385	0.0032	7.60×10^{-5}	6.33×10^{-6}

Limit: Thyroid = 30 rem, Whole Body = 5 rem

Occupational Radiological Exposure Rate from the LEU Core

Distance	Thyroid Dose		Whole Body Dose	
	Rate (rem / hr)	5 Minute Exposure (rem)	Rate (rem / hr)	5 Minute Exposure (rem)
Inside Reactor Building	0.0285	0.0024	5.63×10^{-5}	4.69×10^{-6}

Limit: Thyroid = 30 rem, Whole Body = 5 rem

Radiological Exposure for the Public from HEU Core

Distance (m)	Time of Exposure (hr)	Thyroid Dose (rem)			Whole Body Dose (rem)		
		Leak Rate (%Vol / hr)			Leak Rate (%Vol / hr)		
		10	20	100	10	20	100
16.5	2	0.00181	0.00329	0.00863	1.4×10^{-6}	2.5×10^{-6}	6.6×10^{-6}
190	24	0.000243	0.000300	0.000340	2.2×10^{-7}	2.4×10^{-7}	2.6×10^{-7}

Limit: Thyroid = 3 rem, Whole Body = 0.5 rem

Radiological Exposure for the Public from LEU Core

Distance (m)	Time of Exposure (hr)	Thyroid Dose (rem)			Whole Body Dose (rem)		
		Leak Rate (%Vol / hr)			Leak Rate (%Vol / hr)		
		10	20	100	10	20	100
16.5	2	0.00134	0.00243	0.00639	1.0×10^{-6}	1.8×10^{-6}	4.9×10^{-6}
190	24	0.000180	0.000222	0.000251	1.6×10^{-7}	1.8×10^{-7}	1.9×10^{-7}

Limit: Thyroid = 3 rem, Whole Body = 0.5 rem

The results indicate that the doses from the FHA accident are significantly less than the accidental dose limits which are listed in each table summary. The occupational exposure is given in terms of dose rate in rem per hour and the exposure received over a 5-minute period. A period of 5 minutes is considered to be a conservative time for a worker in the reactor cell to evacuate the cell in event of a fuel handling accident. The detailed results for the occupational exposure analysis are given in Table 13-11 in terms of the dose rate in rem per hour. The detailed results for the public exposure are given in Table 13-12 and Table 13-13 for the maximum exposure and the nearest residence, respectively. The detailed results for the public exposure are given for the assumed leak rate of 20% volume/hr only.

Table 13-11 Thyroid Doses and Whole Body Doses Calculated for the Occupational Exposure for the FHA in the HEU and LEU Cores

	HEU Core		**LEU Core**	
Isotope	**Thyroid Dose, rem/hr**	**Whole Body Dose, rem/hr**	**Thyroid Dose, rem/hr**	**Whole Body Dose, rem/hr**
Kr 85	-	3.61E-08	-	2.67E-08
Kr 85m	-	1.66E-09	-	1.23E-09
Kr 88	-	4.06E-11	-	3.00E-11
I 130	6.85E-09	-	5.05E-09	-
I 131	3.60E-02	-	2.66E-02	-
I 132	2.47E-04	-	1.83E-04	-
I 133	2.30E-03	-	1.70E-03	-
I 135	3.38E-06	-	2.50E-06	-
Xe 133	-	6.61E-05	-	4.89E-05
Xe 133m	-	2.41E-06	-	1.78E-06
Xe 135	-	7.44E-06	-	5.51E-06
Xe 135m	-	3.42E-08	-	2.53E-08
Total Dose	**3.85E-02**	**7.60E-05**	**2.85E-02**	**5.63E-05**

These doses are significantly less than the accidental dose limits of 30 rem to the thyroid and 5 rem to the whole body for occupational exposure at 1 hour. Many hours, up to several hundred, of exposure may occur before the doses would come close to the dose limits. A period of 5 minutes is considered to be a conservative time to evacuate the reactor cell in the event of a fuel handling accident.

Table 13-12 Thyroid Doses and Whole Body Doses Calculated for Public Exposure at 16.5 m for the FHA in the HEU and LEU Cores with 20% Leak Rate

	HEU Core		LEU Core	
Isotope	**Thyroid Dose, rem**	**Whole Body Dose, rem**	**Thyroid Dose, rem**	**Whole Body Dose, rem**
Kr 85	-	4.547E-11	-	3.360E-11
Kr 85m	-	8.538E-11	-	6.310E-11
Kr 88	-	6.165E-12	-	4.557E-12
I 130	5.580E-10	-	4.114E-10	-
I 131	3.077E-03	-	2.277E-03	-
I 132	1.623E-05	-	1.201E-05	-
I 133	1.912E-04	-	1.414E-04	-
I 135	2.632E-07	-	1.947E-07	-
Xe 133	-	2.021E-06	-	1.495E-06
Xe 133m	-	4.084E-08	-	3.021E-08
Xe 135	-	4.333E-07	-	3.208E-07
Xe 135m	-	1.182E-09	-	8.745E-10
Total Dose	**3.285E-03**	**2.497E-06**	**2.431E-03**	**1.847E-06**

Table 13-13 Thyroid Doses and Whole Body Doses Calculated for Public Exposure at 190 m for the FHA in the HEU and LEU Cores with 20% Leak Rate

	HEU Core		LEU Core	
Isotope	**Thyroid Dose, rem**	**Whole Body Dose, rem**	**Thyroid Dose, rem**	**Whole Body Dose, rem**
Kr 85	-	4.694E-12	-	3.468E-12
Kr 85m	-	5.740E-12	-	4.242E-12
Kr 88	-	3.575E-13	-	2.642E-13
I 130	4.449E-11	-	3.280E-11	-
I 131	2.830E-04	-	2.094E-04	-
I 132	8.660E-07	-	6.408E-07	-
I 133	1.616E-05	-	1.195E-05	-
I 135	1.881E-08	-	1.391E-08	-
Xe 133	-	2.043E-07	-	1.511E-07
Xe 133m	-	4.010E-09	-	2.967E-09
Xe 135	-	3.493E-08	-	2.585E-08
Xe 135m	-	4.042E-11	-	2.990E-11
Total Dose	**3.000E-04**	**2.432E-07**	**2.220E-04**	**1.800E-07**

The public doses in Tables 13-12 and 13-13 are extremely small in comparison with the accidental dose limits of 3 rem to the thyroid and 0.5 rem to the whole body for public exposure.

13.4 Maximum Hypothetical Accident (MHA)

The maximum hypothetical accident for the UFTR is a core-crushing accident in which the core is assumed to be severely crushed in either the horizontal or vertical direction by postulating that a 4500 lb concrete shield block is inadvertently dropped onto the core. Based on the design of the facility and the size and weight of the concrete blocks, it is difficult to conceive of how the core would actually be crushed. Nevertheless, even though the possibility of this hypothetical accident is extremely remote, the hypothesis is made that dropping of a 4500 lb concrete shield block would result in severe mechanical damage to the fuel and a significant release of fission products.

Section 4.2.7 of the Technical Specifications was augmented to require at least three days to pass after UFTR power operations before not only fuel handling but also before moving the last two layers of concrete blocks to access the fuel. This new Tech Spec would limit the possible/potential consequences of fuel handling accidents and preclude damaging a fuel bundle with a dropped shield block before three days have elapsed.

The following data and assumptions were used to evaluate the source term associated with this accident:

(1) The reactor is operated at 100 kW steady-state power for 30 days with an equilibrium concentration of fission products.

(2) The fuel elements with highest power in the HEU and LEU cores based on MCNP5 calculations were selected for evaluation. These were element 5-1 with a power of 5.77 kW in the HEU core and bundle 2-3 with a power of 5.45 kW in the LEU core.

These bundle powers were derived from an MCNP tally that assumes that all of the energy produced is deposited locally. This assumption results in maximum bundle powers that are slightly larger than those shown in Table 4-14, which accounts for energy deposited in the fuel plates and gamma energy deposited in the coolant, moderator, and structural materials.

(3) Radioisotope inventories were calculated three days after shutdown from power operation. To ensure that the three day period is satisfied, UFTR has augmented Technical Specification 4.2.7 (1) for the LEU core to require at least three days to pass after UFTR power operations before moving the last two layers of concrete blocks allowing to access the fuel and before fuel handling. This limits the possible/potential consequences of fuel handling accidents and precludes damaging a fuel bundle with a dropped shield block before three days have elapsed. The HEU core was analyzed with the same assumption for comparison purposes.

(4) The radioisotopes of greatest significance for release in case of an accident are the radioiodines and the noble gases, krypton and xenon.

(5) All of the water is assumed to drain out of the core in less than one second, so that any fission product release would be directly to the air of the reactor cell.

(6) It is postulated that the core would undergo severe mechanical damage due to the core crushing accident and that this damage would be sufficient to expose fuel surface areas equivalent to stripping the aluminum cladding from one entire HEU element in the HEU core and one LEU element in the LEU core. It is further assumed (Ref. 19) that 100% of the gaseous activity produced within the recoil range of the particles (1.37×10^{-3} cm) or 2.7% of the total gaseous activity instantaneously escapes from the fuel element into the reactor cell.

13.4.1 Radionuclide Inventories

Radionuclide inventories for the highest power fuel element in the HEU and LEU cores core were calculated using the ORIGEN-S code using the assumptions in the previous paragraph. The activity of the krypton, iodine, and xenon isotopes in the HEU and LEU cores are shown in Table 13-14 along with the inventory (2.7% of the total) that is assumed to escape from the damaged fuel into the air of the reactor cell.

Table 13-14 Calculated Radionuclide Inventories (Ci) Released into the Reactor Cell from the Maximum Hypothetical Accident in the HEU and LEU Cores

	HEU Core		LEU Core	
Isotope	**Ci in One Fuel Element Three Days after Shutdown**	**2.7% of Previous Column**	**Ci in One Fuel Element Three Days after Shutdown**	**2.7% of Previous Column**
Kr 85	7.75E-02	2.09E-03	7.29E-02	1.97E-03
Kr 85m	9.28E-04	2.51E-05	8.73E-04	2.36E-05
Kr 88	4.10E-06	1.11E-07	3.85E-06	1.04E-07
I 129	8.58E-08	2.32E-09	8.09E-08	2.18E-09
I 130	2.07E-04	5.60E-06	1.94E-04	5.23E-06
I 131	104.8	2.83	98.7	2.67
I 132	116.1	3.13	109.3	2.95
I 133	30.9	0.833	29.0	0.784
I 135	0.157	4.23E-03	0.148	3.99E-03
Xe 133	259.3	7.00	244.2	6.59
Xe 133m	5.64	0.152	5.31	0.143
Xe 135	4.24	0.115	4.01	0.108
Xe 135m	2.56E-02	6.92E-04	2.41E-02	6.51E-04

13.4.2 Methodology for Dose Calculations

The methodology that was used for the Maximum Hypothetical Accident is the same as described in Section 13.3.2 for the Fuel-Handling Accident. Note the I-129 isotope was not considered in the subsequent calculations because the dose amount is extremely small as indicated in Table 13-14.

13.4.3 Dose Calculations for Maximum Hypothetical Accident

The calculated thyroid doses and whole body doses for the occupational and public exposures for the Maximum Hypothetical Accident are shown in Table 13-15.

The results indicate the doses from the MHA accident are less than the accidental dose limits which are listed in each table summary. The occupational exposure is given in terms of dose rate in rem per hour and the exposure received over a 5-minute period. A period of 5 minutes is considered to be a conservative time for a worker in the reactor cell to evacuate the cell in the event that the maximum hypothetical accident would occur. The detailed results for the occupational exposure analysis are given in Table 13-16 in terms of the dose rate in rem per hour. The detailed results for the public exposure are given in Table 13-17 and Table 13-18 for the maximum exposure and the nearest residence, respectively. The detailed results for the public exposure are given for the assumed leak rate of 20% volume/hr only.

Table 13-15 Summary of Dose Results for the MHA in the HEU and LEU Cores

Occupational Radiological Exposure Rate from the HEU Core

Distance	Thyroid Dose		Whole Body Dose	
	Rate (rem / hr)	5 Minute Exposure (rem)	Rate (rem / hr)	5 Minute Exposure (rem)
Inside Reactor Building	2.47	0.206	0.0048	0.0004

Limit: Thyroid = 30 rem, Whole Body = 5 rem

Occupational Radiological Exposure Rate from the LEU Core

Distance	Thyroid Dose		Whole Body Dose	
	Rate (rem / hr)	5 Minute Exposure (rem)	Rate (rem / hr)	5 Minute Exposure (rem)
Inside Reactor Building	2.33	0.194	0.0045	0.00038

Limit: Thyroid = 30 rem, Whole Body = 5 rem

Radiological Exposure for the Public from HEU Core

Distance (m)	Time of Exposure (hr)	Thyroid Dose (rem)			Whole Body Dose (rem)		
		Leak Rate (%Vol / hr)			Leak Rate (%Vol / hr)		
		10	20	100	10	20	100
16.5	2	0.116	0.211	0.554	$8.3x10^{-5}$	$1.5x10^{-4}$	$4.0x10^{-4}$
190	24	0.0157	0.0193	0.0218	$1.3x10^{-5}$	$1.5x10^{-5}$	$1.6x10^{-5}$

Limit: Thyroid = 3 rem, Whole Body = 0.5 rem

Radiological Exposure for the Public from LEU Core

Distance (m)	Time of Exposure (hr)	Thyroid Dose (rem)			Whole Body Dose (rem)		
		Leak Rate (%Vol / hr)			Leak Rate (%Vol / hr)		
		10	20	100	10	20	100
16.5	2	0.109	0.199	0.522	$7.8x10^{-5}$	$1.4x10^{-4}$	$3.7x10^{-4}$
190	24	0.0193	0.0182	0.0205	$1.3x10^{-5}$	$1.4x10^{-5}$	$1.5x10^{-5}$

Limit: Thyroid = 3 rem, Whole Body = 0.5 rem

Table 13-16 Thyroid Doses and Whole Body Doses Calculated for the Occupational Exposure for the MHA in the HEU and LEU Cores.

	HEU Core		LEU Core	
Isotope	**Thyroid Dose, rem/hr**	**Whole Body Dose, rem/hr**	**Thyroid Dose, rem/hr**	**Whole Body Dose, rem/hr**
Kr 85	-	2.41E-06	-	2.27E-06
Kr 85m	-	3.86E-08	-	3.63E-08
Kr 88	-	7.28E-10	-	6.85E-10
I 130	3.16E-07	-	2.95E-07	-
I 131	2.34	-	2.21	-
I 132	1.55E-02	-	1.46E-02	-
I 133	1.15E-01	-	1.08E-01	-
I 135	1.02E-04	-	9.57E-05	-
Xe 133	-	4.33E-03	-	4.08E-03
Xe 133m	-	1.48E-04	-	1.40E-04
Xe 135	-	2.66E-04	-	2.51E-04
Xe 135m	-	1.03E-06	-	9.70E-07
Total Dose	**2.47**	**4.75E-03**	**2.33**	**4.48E-03**

These doses are less than the accidental dose limits of 30 rem to the thyroid and 5 rem to the whole body for occupational exposure at 1 hour. Several hours of exposure may occur before the doses would become close to the dose limits. A period of 5 minutes is considered to be a conservative time to evacuate the reactor cell in the event that the maximum hypothetical accident should occur.

Table 13-17 Thyroid Doses and Whole Body Doses Calculated for Public Exposure at 16.5 m for the MHA in the HEU and LEU Cores with 20% Leak Rate

	HEU Core		LEU Core	
Isotope	**Thyroid Dose, rem**	**Whole Body Dose, rem**	**Thyroid Dose, rem**	**Whole Body Dose, rem**
Kr 85	-	3.036E-09	-	2.855E-09
Kr 85m	-	1.988E-09	-	1.870E-09
Kr 88	-	1.106E-10	-	1.040E-10
I 130	2.573E-08	-	2.402E-08	-
I 131	2.004E-01	-	1.887E-01	-
I 132	1.016E-03	-	9.573E-04	-
I 133	9.564E-03	-	9.003E-03	-
I 135	7.916E-06	-	7.454E-06	-
Xe 133	-	1.325E-04	-	1.248E-04
Xe 133m	-	2.514E-06	-	2.367E-06
Xe 135	-	1.550E-05	-	1.464E-05
Xe 135m	-	3.555E-08	-	3.348E-08
Total Dose	**0.2110**	**1.505E-04**	**0.1987**	**1.418E-04**

Table 13-18 Thyroid Doses and Whole Body Doses Calculated for Public Exposure at 190 m for the MHA in the HEU and LEU Cores with 20% Leak Rate

	HEU Core		LEU Core	
Isotope	**Thyroid Dose, rem**	**Whole Body Dose, rem**	**Thyroid Dose, rem**	**Whole Body Dose, rem**
Kr 85	-	3.134E-10	-	2.948E-10
Kr 85m	-	1.337E-10	-	1.257E-10
Kr 88	-	6.412E-12	-	6.032E-12
I 130	2.051E-09	-	1.915E-09	-
I 131	1.843E-02	-	1.735E-02	-
I 132	5.423E-05	-	5.108E-05	-
I 133	8.080E-04	-	7.607E-04	-
I 135	5.656E-07	-	5.327E-07	-
Xe 133	-	1.339E-05	-	1.261E-05
Xe 133m	-	2.469E-07	-	2.324E-07
Xe 135	-	1.249E-06	-	1.180E-06
Xe 135m	-	1.216E-09	-	1.145E-09
Total Dose	**1.929E-02**	**1.489E-05**	**1.816E-02**	**1.402E-05**

The public doses in Tables 13-17 and 13-18 are small in comparison with the accidental dose limits of 3 rem to the thyroid and 0.5 rem to the whole body for public exposure.

14. Technical Specifications

For the UFTR HEU to LEU conversion, the only changes required for the UFTR Technical Specifications involve the fuel type and certain related specifications.

First, on page 4 of the Tech Specs, in Section 2.1, Safety Limits, specifications (1), (2) and (3), the safety limits on power level, primary coolant flow rate, and primary coolant outlet temperature from any fuel box are changed from their current specifications quoted as follows:

(1) The steady-state power level shall not exceed 100 kWt.
(2) The primary coolant flow rate shall be greater than 18 gpm at all power levels greater than 1 watt.
(3) The primary coolant outlet temperature from any fuel box shall not exceed 200° F.

to new specifications on power level, flow rate, and primary coolant outlet temperature from any fuel box, correlated with the existing limiting safety system setting (LSSS) (trip points) on power level of 125 kW, flow rate of 30 gpm and primary coolant outlet temperature of 200° F and the accident analysis results presented to assure conservative limits in Section 4.7 as follows:

(1) The power level shall not exceed 190 kW.
(2) The primary coolant flow rate shall be greater than 23 gpm at all power levels greater than 1 watt.
(3) The primary coolant outlet temperature from any fuel box shall not exceed 160° F.

As noted in the Section 4.7 analyses, the three parameters of power level, flow rate and primary coolant outlet temperature are interdependent so the safety limits are based on nominal as well as conservative analyses. For the nominal analyses, any two parameters are varied from nominal operating conditions to reach onset of nucleate boiling in the LEU core. In the conservative approach, any two parameters are varied from the LSSS point to reach onset of nucleate boiling. The actual proposed safety limits are based on a linear average of the two approaches as detailed in Section 4.7. In addition, the steady-state reference in specification (1) is removed as not applicable and the change from kWt to kW in specification (1) is simply to be consistent with the remainder of the Tech Specs. The resulting bases for specifications (1), (2) and (3) are then addressed together after the specifications in Section 2.1 as all three are interdependent with the objective now to prevent onset of nucleate boiling as a conservative objective and, as previously, to assure the fuel remains below temperatures at which fuel degradation would occur.

Second, on page 6 of the Tech Specs, in Section 3.1, Reactivity Limitations, paragraph (2), the core excess reactivity at cold critical, without xenon poisoning, is changed from not exceeding 2.3% $\Delta k/k$ to not exceeding 1.4% $\Delta k/k$, again based on the accident analysis results presented in Section 13 and considering the actual realistic excess reactivity needed for operations.

Third, on page 13 of the Tech Specs, in Section 3.5, Limitations on Experiments, paragraph (3)(b), the limit on total absolute reactivity worth of all experiments is changed from not exceeding 2.3% $\Delta k/k$ to not exceeding 1.4% $\Delta k/k$ to be consistent with the change made on overall reactivity limitations per the previous paragraph.

Fourth, on page 15 of the Tech Specs, in Section 3.7, Fuel and Fuel Handling, paragraph (1), the description of fuel elements is changed from "fuel elements consisting of 11 plates each . . ." to "fuel elements consisting of 14 plates each . . . ." This change is necessitated by the basic LEU fuel assembly design selected for the conversion as described in Section 4 for the LEU fuel.

Fifth, on page 23 of the Tech Specs, in Section 5.3, Reactor Fuel, in the first paragraph, line 2, the enrichment is changed to specify "no more than about 19.75% U-235" based on the LEU fuel selections. In lines 4 through 6, the allowable fabrication methodology is changed to allow high purity uranium silicide-aluminum dispersion fuel in addition to the currently allowed high purity aluminum-uranium alloy. In the last line of the paragraph, the loading of U-235 per plate is changed to "nominally 12.5 g of U-235 per fuel plate." Again, these specifications are in agreement with the analysis provided in Section 4 for the LEU fuel.

Sixth, on page 23 of the Tech Specs, in Section 5.4, Reactor Core, in the first paragraph, in line 1, the number of plates per assembly becomes 14 for LEU bundles versus 11 for HEU bundles. Similarly, in line 4, a full assembly shall be replaced with no fewer than 13 plates in a pair of partial assemblies versus 10 plates for the HEU core. Finally, in the second paragraph, the table giving the required nominal fuel element specifications is updated to provide the parameters for the LEU fuel per the analysis summarized in Section 4.

Table 14-1 Summary of LEU Fuel Parameters to be Updated in the Technical Specifications

Item	**Specification**
Overall size (bundle)	2.845 in. x 2.26 in. x 25.6 in.
Clad thickness	0.015 in.
Plate thickness	0.050 in.
Water channel width	0.111 in.
Number of plates	Standard fuel element – 14 fueled plates; Partial element – no fewer than 11 plates in a pair of partial assemblies
Plate attachment	Bolted with spacers
Fuel content per plate	12.5 g U-235 nominal

15. Other License Considerations

The necessary documentation of reactor parameters and status will be provided after the conversion from HEU to LEU fuel.

Appendix A

A.1 Determination of Material Composition

This section includes information on material composition, and discusses the methodology used to accurately determine isotopic concentrations for the depleted cores.

Table A.1-1 presents the fuel concentrations for the HEU fresh core.

Table A.1-1 UFTR HEU Core Fuel Loading at Beginning-of-Life

[redacted]

Table A.1-2 presents the power peak-to-average ratios that are used for the determination of material concentrations in the HEU core.

Table A.1-2 HEU Power Peak-to-Average Ratio for Core at Beginning-of-Life

Bundle Number	Axial Segment	Peak to Average Ratio	Bundle Number	Axial Segment	Peak to Average Ratio
1-1	1	4.26E-01	4-1	1	1.01E+00
1-1	2	5.44E-01	4-1	2	1.27E+00
1-1	3	4.02E-01	4-1	3	9.57E-01
1-2	1	9.18E-01	4-2	1	1.15E+00
1-2	2	1.14E+00	4-2	2	1.42E+00
1-2	3	8.36E-01	4-2	3	1.05E+00
1-3	1	8.89E-01	4-3	1	9.21E-01
1-3	2	1.12E+00	4-3	2	1.15E+00
1-3	3	8.40E-01	4-3	3	8.59E-01
1-4	1	1.03E+00	4-4	1	1.05E+00
1-4	2	1.27E+00	4-4	2	1.30E+00
1-4	3	9.55E-01	4-4	3	9.46E-01
2-1	1	9.86E-01	5-1	1	1.20E+00
2-1	2	1.22E+00	5-1	2	1.46E+00
2-1	3	8.87E-01	5-1	3	1.07E+00
2-2	1	9.62E-01	5-2	1	1.14E+00
2-2	2	1.19E+00	5-2	2	1.38E+00
2-2	3	8.68E-01	5-2	3	9.37E-01
2-3	1	1.11E+00	5-3	1	1.08E+00
2-3	2	1.37E+00	5-3	2	1.32E+00
2-3	3	1.02E+00	5-3	3	9.40E-01
2-4	1	1.08E+00	5-4	1	1.03E+00
2-4	2	1.33E+00	5-4	2	1.26E+00
2-4	3	9.89E-01	5-4	3	8.30E-01
3-1	1	8.17E-01	6-1	1	9.35E-01
3-1	2	1.02E+00	6-1	2	1.13E+00
3-1	3	7.44E-01	6-1	3	7.63E-01
3-3	1	8.96E-01	6-2	1	8.10E-01
3-3	2	1.11E+00	6-2	2	1.01E+00
3-3	3	8.31E-01	6-2	3	7.38E-01
3-4	1	7.85E-01	6-3	1	8.58E-01
3-4	2	9.82E-01	6-3	2	1.05E+00
3-4	3	7.45E-01	6-3	3	6.89E-01

The masses of the fuel matrix, impurities in the silicide, and impurities in Al are given in Tables A.1-3a to A.1-3c.

Table A.1-3a Weights of major elements and isotopes in a LEU fuel plate

Isotope	Mass per fuel plate (g)
U-234	1.03E-01
U-235	[redacted]
U-236	6.57E-02
U-238	5.08E+01
Si	5.00E+00
Al	3.23E+01

Table A.1-3b Impurities in U_3Si_2 Powder

Isotope	Concentration (ppm)	Mass (per Gram Fuel Meat)
Al	131.67	8.95E-05
Ba	2.00	1.36E-06
Be	0.50	3.40E-07
B	1.33	1.82E-07
Cd	0.50	3.40E-07
Ca	20.00	1.36E-05
C	244.00	1.66E-04
Cr	18.33	1.25E-05
Co	5.00	3.40E-06
Cu	100.83	6.85E-05
Eu	0.20	1.36E-07
Gd	0.20	1.36E-07
Fe	608.50	4.13E-04
Pb	0.50	3.3974E-07
Li	0.10	6.80E-08
Mg	10.00	6.80E-06
Mn	8.67	5.89E-06
Mo	3.00	2.04E-06
Ni	43.33	2.94E-05
N	55.00	3.74E-05
P	20.00	1.36E-05
Sm	0.20	1.36E-07
Ag	1.00	6.79E-07
Na	10.00	6.79E-06
Sn	1.00	6.79E-07
W	21.67	1.47E-05
V	4.50	3.06E-06
Zn	20.00	1.36E-05
Zr	3.83	2.60E-06

Table A.1-3c Impurities in Aluminum Powder Used in Fuel

Isotope	Mass Fraction (wt %)	Mass (per Gram Fuel Meat)
Zn	0.02	6.41E-05
Cu	0.001	3.21E-06
Cd	0.001	3.21E-06
Li	0.001	3.21E-06
B	0.001	3.21E-06
Fe	0.167	5.35E-04
O	0.097	3.11E-04

It is also important to mention that an effort was made in using realistic local parameters. Using coolant temperature profiles in the UFTR FSAR, a power shape was constructed to obtain an initial axial power peaking function for a generic fuel bundle. Average fuel and cladding temperatures as well as coolant temperature and density were calculated and used in an initial core physics calculation. The resulting power peak-to-average ratios where used to recalculate more accurate local parameters for each fuel bundle. Three average axial fuel temperatures are used for the whole core while an average coolant density per fuel box is considered. Tables A.1-4 and A.1-5 present the data local parameters for the HEU and LEU, respectively.

Table A.1-4 UFTR LEU Core Average Coolant Densities per Fuel Box

Coolant Box	Average Coolant Density Across Bundle Region (g/cc)
Box 1	0.99395
Box 2	0.99373
Box 3	0.99419
Box 4	0.99409
Box 5	0.99397
Box 6	0.99462

Table A.1-5 UFTR LEU Average Fuel Temperature

Axial Segment [1]	Average Fuel Temperature (K)
1	309.32
2	316.74
3	316.15

[1] Each axial segment represents a third of the bundle length with segment #1 being at the bottom

The peak-to-average power ratios for the LEU core are given in Table A.1-6.

Table A.1-6 LEU Peak-to-Average Power Ratio for Core at Beginning-of-Life

Bundle Number	Axial Segment	Peak to Average Ratio	Bundle Number	Axial Segment	Peak to Average Ratio
1-1	1	8.87E-01	4-1	1	9.14E-01
1-1	2	1.11E+00	4-1	2	1.14E+00
1-1	3	8.05E-01	4-1	3	8.49E-01
1-2	1	1.03E+00	4-2	1	1.05E+00
1-2	2	1.26E+00	4-2	2	1.28E+00
1-2	3	8.49E-01	4-2	3	9.27E-01
1-3	1	9.68E-01	4-3	1	8.19E-01
1-3	2	1.21E+00	4-3	2	1.03E+00
1-3	3	9.00E-01	4-3	3	7.46E-01
1-4	1	1.12E+00	4-4	1	9.36E-01
1-4	2	1.38E+00	4-4	2	1.15E+00
1-4	3	9.93E-01	4-4	3	7.77E-01
2-1	1	1.10E+00	5-1	1	1.10E+00
2-1	2	1.34E+00	5-1	2	1.34E+00
2-1	3	8.89E-01	5-1	3	9.51E-01
2-2	1	1.09E+00	5-2	1	1.06E+00
2-2	2	1.33E+00	5-2	2	1.28E+00
2-2	3	8.80E-01	5-2	3	8.85E-01
2-3	1	1.20E+00	5-3	1	9.82E-01
2-3	2	1.47E+00	5-3	2	1.19E+00
2-3	3	1.05E+00	5-3	3	7.95E-01
2-4	1	1.19E+00	5-4	1	9.41E-01
2-4	2	1.46E+00	5-4	2	1.14E+00
2-4	3	1.04E+00	5-4	3	7.51E-01
3-1	1	9.57E-01	6-1	1	8.78E-01
3-1	2	1.17E+00	6-1	2	1.07E+00
3-1	3	7.88E-01	6-1	3	7.41E-01
3-2	1	6.21E-01	6-2	1	7.70E-01
3-2	2	7.85E-01	6-2	2	9.54E-01
3-2	3	5.67E-01	6-2	3	7.12E-01
3-3	1	1.04E+00	6-3	1	7.86E-01
3-3	2	1.27E+00	6-3	2	9.63E-01
3-3	3	9.18E-01	6-3	3	6.41E-01
3-4	1	8.78E-01			
3-4	2	1.10E+00			
3-4	3	8.16E-01			

Note that, as in the FSAR, the average moderator temperature is assumed to be equal to the average coolant temperature (307.8K). To examine the validity of this assumption, we utilize the following relation to estimate the operating time required to reach the assumed temperature,

$$mC_p(T - T_{init}) = P \times t \qquad \text{(A.1.1)}$$

where m is the mass of graphite, C_p is specific heat of graphite, T is the temperature, P is the core power and t is the time of operation at power P. To solve for t, we determine the deposited power P (3.69E+03 ± 0.0002 J/s) in graphite at full power using MCNP5 (in neutron-photon mode for a more accurate energy deposition), assume an initial temperature of 298K, and a graphite heat capacity of 0.711 J/g/K. Based on these parameters, this initial scoping calculation estimates that this assumption corresponds to the temperature reached after 2 hours of operation at full power. We believe this constitutes an acceptable assumption.

A.2 Power History for the HEU Core

The power history given in Table A.2-1 includes a 1 kW-hr run at the end to account for any short lived isotopes generated during the experiment. The current experiments were run after the reactor was shut down for 5 days and therefore no power history was included for the days leading up to the experiment.

Table A.2-1 Power History for the 20 Oldest Bundles

Year	kW-hr	Full Power Days	Down time in Days	Year	kW-hr	Full Power Days	Down time in Days
1982	14480	1.51	89.74	1994	27599	11.50	353.50
1983	47287	19.70	345.30	1995	21347	8.89	356.11
1984	35879	14.95	350.05	1996	16904	7.04	357.96
1985	19288	8.04	356.96	1997	11615	4.84	360.16
1986	29749	12.40	352.60	1998	3429	1.43	363.57
1987	26677	11.12	353.88	1999	19387	8.08	356.92
1988	35199	14.67	350.33	2000	21744	9.06	355.94
1989	24700	10.29	354.71	2001	11173	4.66	360.34
1990	17519	7.30	357.70	2002	10761	4.48	360.52
1991	21904	9.13	355.87	2003	14536	6.06	358.94
1992	33943	14.14	350.86	2004	14995	6.25	448.75
1993	28798	12.00	353.00	Exp.	1	0.0004	n/a

Note that bundle UF-40 (bundle1-1 in the model) is a half-bundle which was added in 1986 and UF-99 (bundle 3-1 in the model) is a full bundle replacement added in 1990. These bundles have different power histories as shown in Tables A.2-2 and A.2-3.

Table A.2-2 Power History for Bundle UF-40

Year	kW-hr	Full Power Days	Down time in Days	Year	kW-hr	Full Power Days	Down time in Days
1986	22311.75	9.30	264.45	1996	16904	7.04	357.96
1987	26677	11.12	353.88	1997	11615	4.84	360.16
1988	35199	14.67	350.33	1998	3429	1.43	363.57
1989	24700	10.29	354.71	1999	19387	8.08	356.92
1990	17519	7.30	357.70	2000	21744	9.06	355.94
1991	21904	9.13	355.87	2001	11173	4.66	360.34
1992	33943	14.14	350.86	2002	10761	4.48	360.52
1993	28798	12.00	353.00	2003	14536	6.06	358.94
1994	27599	11.50	353.50	2004-	14995	6.25	448.75
1995	21347	8.89	356.11	Exp.	1	0.0004	n/a

Table A.2-3 Power History For Bundle UF-99

Year	kW-hr	Full Power Days	Down time in Days	Year	kW-hr	Full Power Days	Down time in Days
1990	8759.5	3.65	178.85	1998	3429	1.43	363.57
1991	21904	9.13	355.87	1999	19387	8.08	356.92
1992	33943	14.14	350.86	2000	21744	9.06	355.94
1993	28798	12.00	353.00	2001	11173	4.66	360.34
1994	27599	11.50	353.50	2002	10761	4.48	360.52
1995	21347	8.89	356.11	2003	14536	6.06	358.94
1996	16904	7.04	357.96	2004-	14995	6.25	448.75
1997	11615	4.84	360.16	Exp.	1	0.0004	n/a

A.3 Impact of the Boron Content for the HEU Core

Due to the uncertainties in the concentrations of certain impurities, it is necessary to perform a small sensitivity study. Table A.3-1 presents changes in k_{eff} obtained for different concentrations boron-equivalent impurities for the HEU core.

Table A.3-1 Impact of Impurities on the Excess Reactivity of the HEU Core

Case (ppm of natural boron-equivalent)	$\Delta k/k$ (%) [1]
4 ppm in graphite	0.303
6 ppm in graphite	-0.133
0 ppm in cladding	0.254
20 ppm in cladding	-0.177
0 ppm in Al structure	0.146
20 ppm in Al structure	-0.015
Graphite/cladding/structure impurities at minimum	0.546
Graphite/cladding/structure impurities at maximum	-0.472
5.72ppm [2] in fuel aluminum alloy	-0.107

[1] The 1σ relative errors for these values is below 0.00025

[2] This value is taken from ANL intra-laboratory memo of June 30th, 2005

Among the tested parameters in above table, the consideration of the impurities in the fuel aluminum alloy compensate for the observed difference in the core excess reactivity. Note that using the 5.72ppm of natural boron-equivalent impurity in the fuel, the k_{eff} of the depleted core with the control blades at their critical positions is 0.99993 (+/-0.00013).

A.4 Determination of the Critical LEU Core

We have prepared seven LEU fuel configurations to investigate the necessary number of fuel bundles and plates. The seven cases are:

24 bundles,
23 bundles,
22.5 bundles with half dummy bundle located at SE,
22.5 bundles with half dummy bundle located at NE,
22 bundles,

21.5 bundles (same as the current HEU core), and
22 bundles with 10 fuel plates at SE.

Figure A.4-1 below shows the variation of the K_{eff} for different configurations. The selected configuration based on this study is case 7 that includes 22 full bundles and a partially filled bundle with 10 fuel plates and 4 dummy aluminum plates. This case leads to a Keff of 1.00934±.000663 with an excess reactivity 0.925 %. This is very close to the excess reactivity of the HEU core with fresh fuel. Note that because of existing uncertainties in material composition, besides case 7, we have analyzed cases 2 and 3 (with 23 and 22.5 fuel bundles), and estimated the effect of variations in the material impurities. Further detail on the results is provided in Table A.4-1.

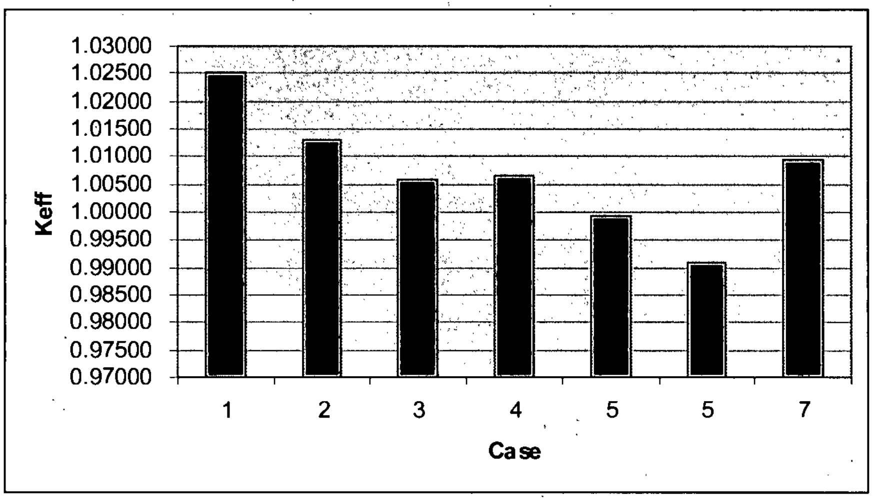


Figure A.4-1 Excess Reactivity for Different LEU Core (1-σ standard deviation is < 0.0007)

Table A.4-1 Comparison of the excess reactivity for different LEU Configurations

Case	Number of Bundles	Description	Blades Positions	K_{eff}	1-σ Standard Deviation
1	24	all fuel bundles	all out (at 47.5 degree)	1.02510	0.000660
2	23	one dummy bundle at SE fuel box, (bundle number 3-2)	all out (at 47.5 degree)	1.01307	0.000600
3	22.5	one dummy bundle at NE fuel box (6-4); half dummy half fuel bundle at SE(3-2)	all out (at 47.5 degree)	1.00601	0.000620
4	22.5	one dummy bundle at SE fuel box (3-2); half dummy half fuel bundle at NE(6-4)	all out (at 47.5 degree)	1.00649	0.000620
5	22	two dummy bundles at NE (6-4) and SE (3-2)	all out (at 47.5 degree)	0.99936	0.000640
6	21.5	bundle layout same as HEU	all out (at 47.5 degree)	0.99076	0.000650
7	22b +10p	one dummy bundle at NE fuel box (6-4); 4 dummy plates 10 fuel plates for bundle at SE(3-2)	all out (at 47.5 degree)	1.00934	0.000630

It is clear that case 7 is the best case, considering the necessary excess reactivity.

A.5 Detailed Flux Profiles for the HEU and LEU cores

In order to tally the axial and radial neutron flux profiles, it is necessary to determine an energy group structure. First, it is useful to evaluate then range of the "thermal" energy group, i.e., the energy group where the neutrons are in thermal equilibrium with their environment and follow a Maxwellian distribution. The Maxwellian distribution is often expressed in term of neutron speed, as given by,

$$4\pi\left(\frac{1}{2\pi c}\right)^{3/2} v^2 e^{-v^2/2c} \tag{A.5.1}$$

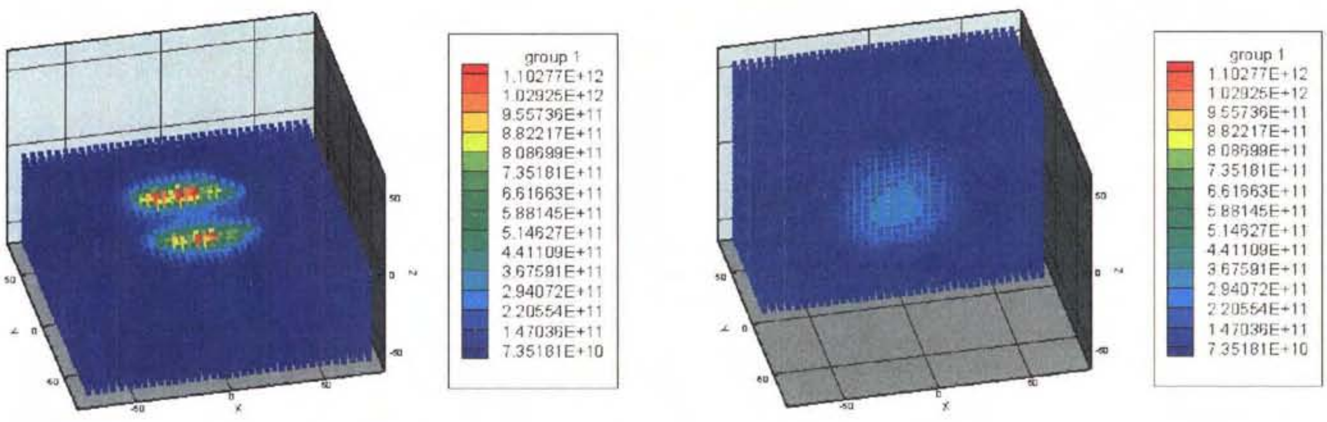
where v represents the neutron speed, c is kT/m, k is the Boltzmann constant, T is the temperature of the gas and m is the mass of the neutron. Using three standard deviations from the average, we evaluate the "maximum" range of the distribution. This range for the UFTR is equal to 0.175eV. Another group boundary is set to 1eV since it is often used as the thermal boundary for reactors. The "epithermal" region is divided into three energy groups, and the "fast" region into two energy groups. Table A.5-1 gives the energy group upper boundaries.

Table A.5-1 Flux Energy Group Structure for UFTR

Spectrum Region	Group Number	Upper Energy (MeV)
Fast	1	2.00e1
	2	1.00e0
Epithermal	3	1.00e-1
	4	1.00e-2
	5	1.00e-4
Thermal	6	1.00e-6
	7	1.75e-7

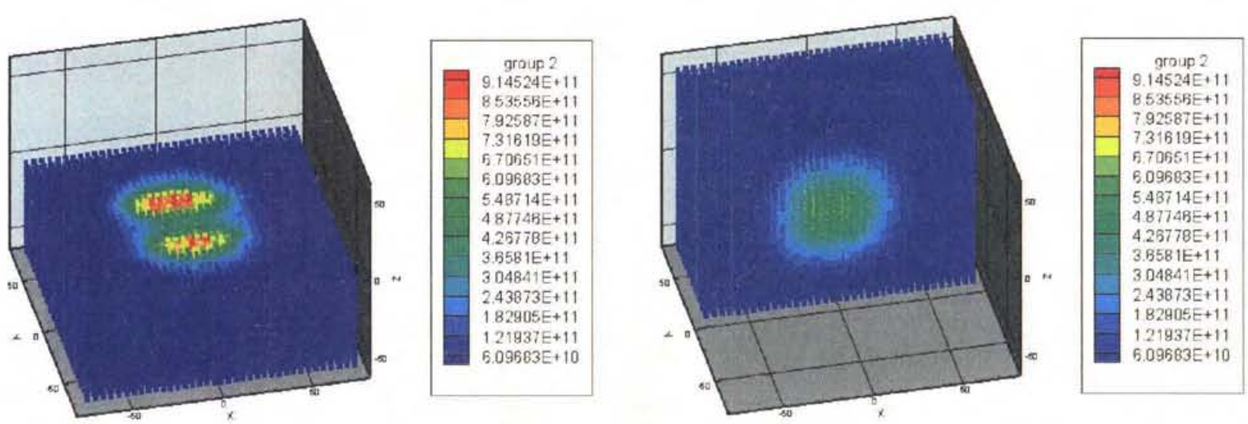
Figures A.5-1 to A.5-7 show the detailed flux profiles for the HEU core obtained for each of the groups listed in Table A.5-1, while flux profiles for the LEU core are presented in Figures A.5-8 to A.5-13.

A.5.1 HEU Detailed Flux Profiles



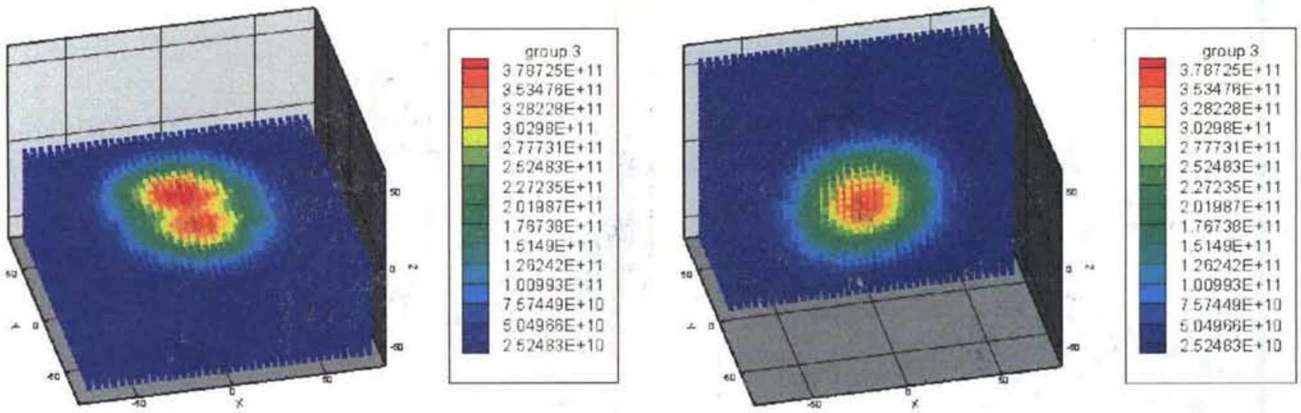
(a) x-y projection **(b) x-z projection**

Figure A.5-1 HEU Flux Distribution for Energy Group 1 (1.0 – 20.0 MeV)



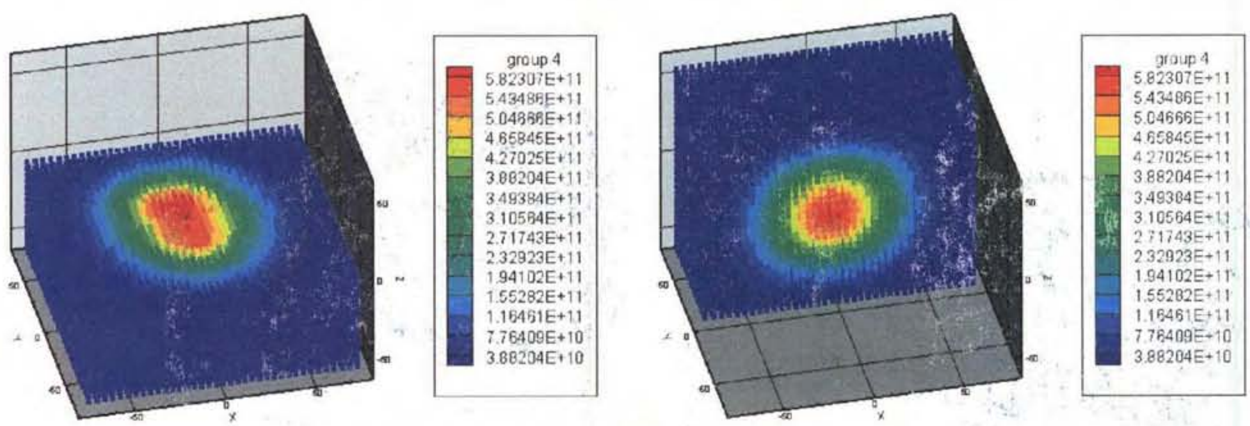
(a) x-y projection **(b) x-z projection**

Figure A.5-2 HEU Flux Distribution for Energy Group 2 (0.10 – 1.0 MeV)



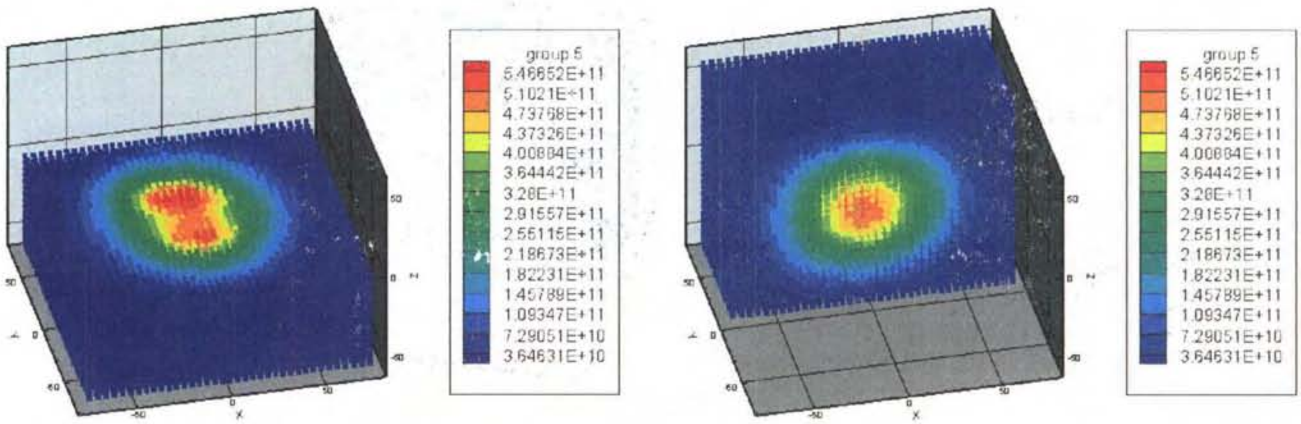
(a) x-y projection **(b) x-z projection**

Figure A.5-3 HEU Flux Distribution of Energy Group 3 (0.01 – 0.10 MeV)



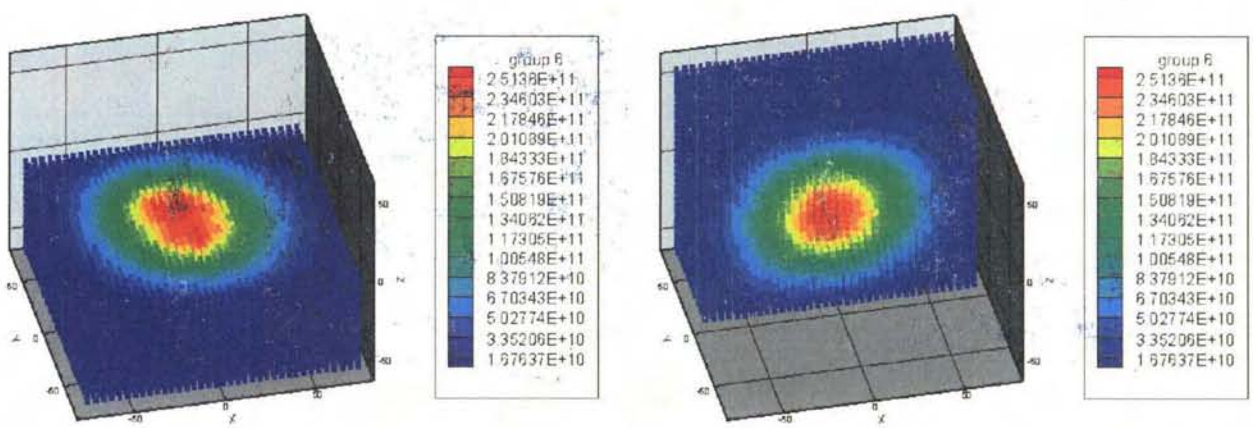
(a) x-y projection **(b) x-z projection**

Figure A.5-4 HEU Flux Distribution for Energy Group 4 (1.0 E-4 - .01 MeV)



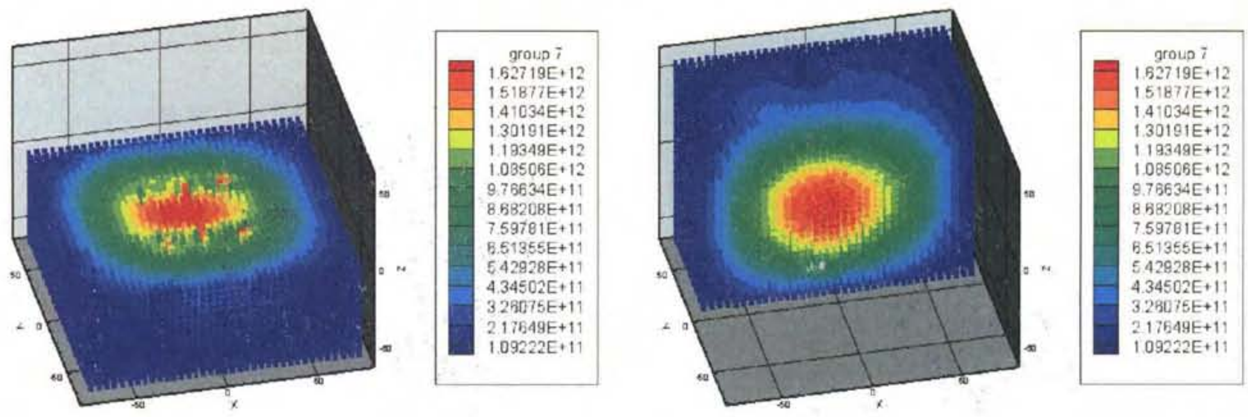
(a) x-y projection **(b) x-z projection**

Figure A.5-5 HEU Flux Distribution for Energy Group 5 (1.0E-6 – 1.0E-4 MeV)



(a) x-y projection **(b) x-z projection**

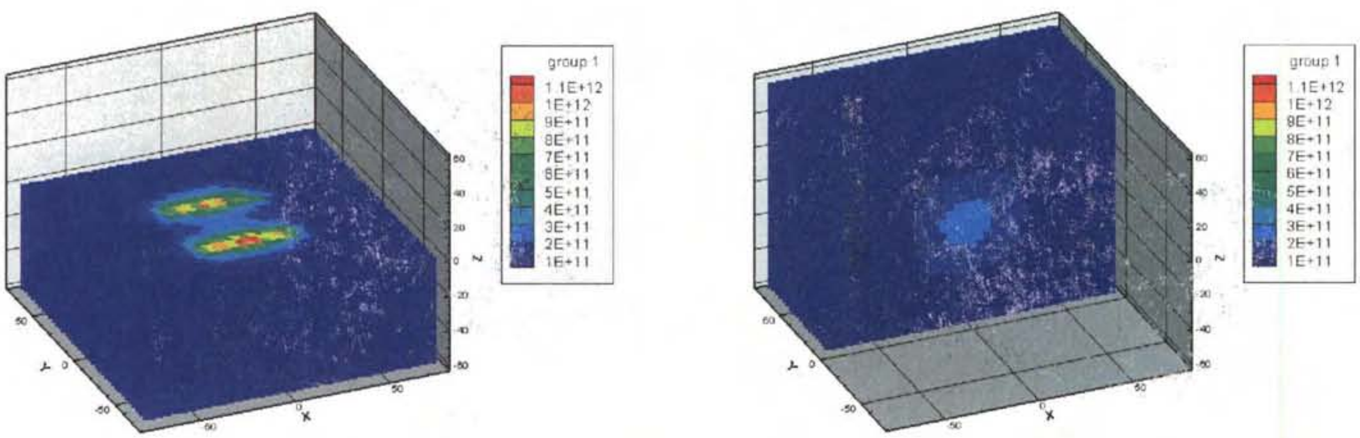
Figure A.5-6 HEU Flux Distribution for Energy Group 6 (1.0E-7 – 1.0E-6 MeV)



(a) x-y projection **(b) x-z projection**

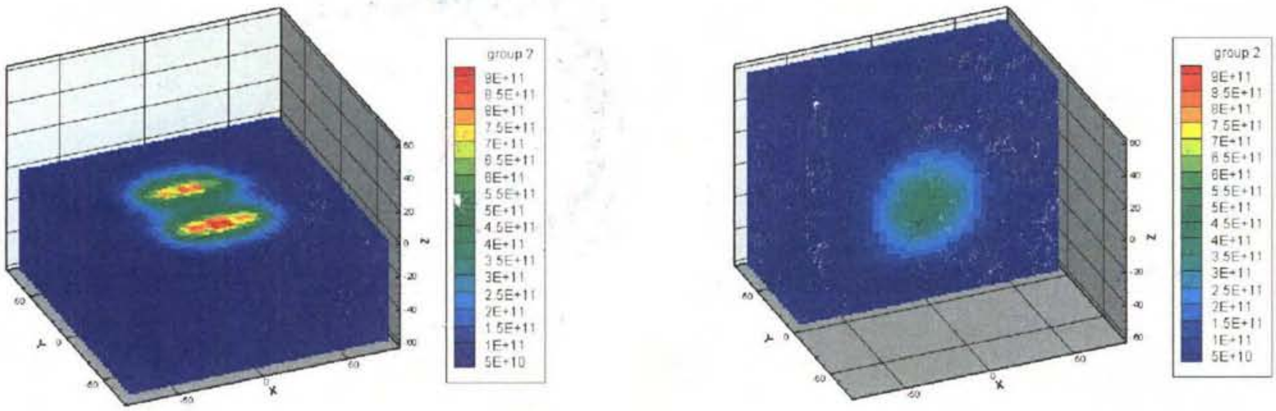
Figure A.5-7 HEU Flux Distribution for Energy Group 7 (0.0 -1.0E-7 MeV)

A.5.2 LEU Detailed Flux Profiles



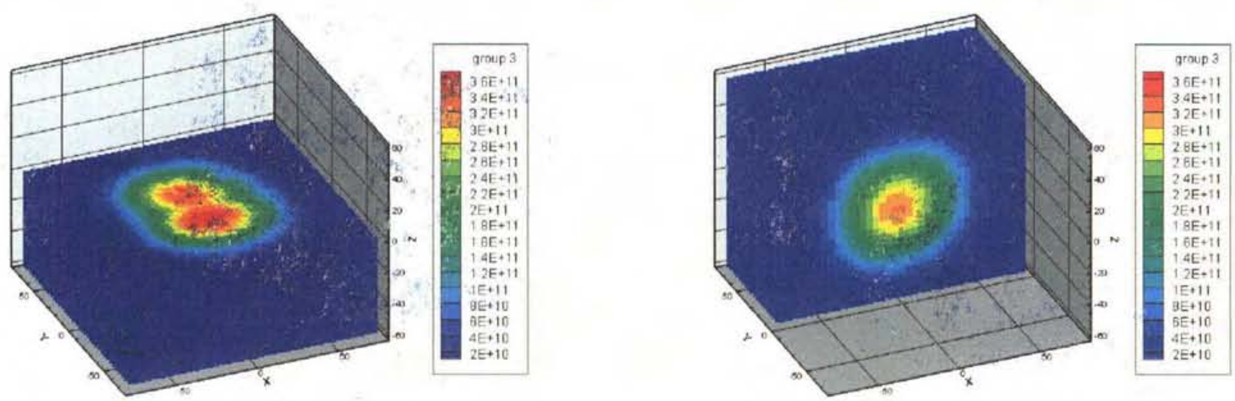
(a) x-y projection **(b) x-z projection**

Figure A.5-8 LEU Flux Distribution for Energy Group 1 (1.0 – 20.0 MeV)



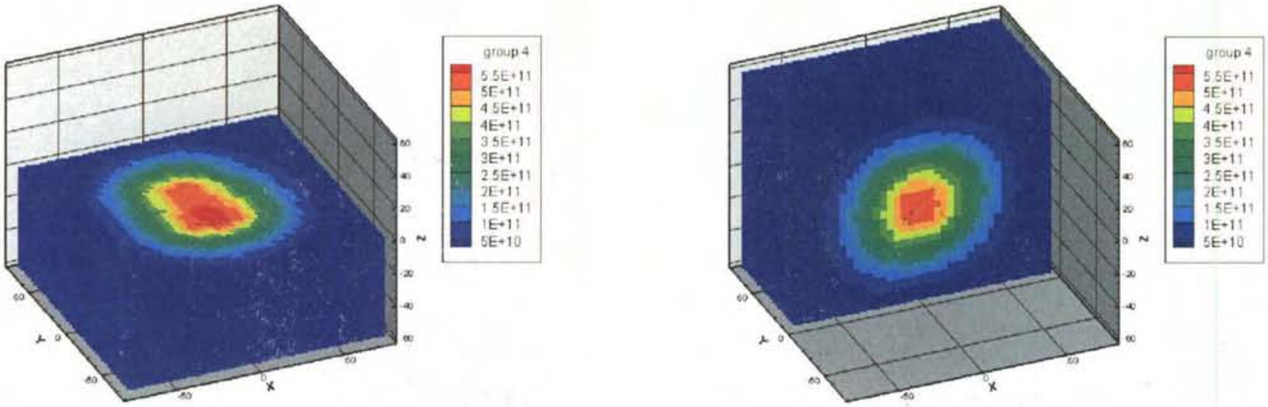
(a) x-y projection **(b) x-z projection**

Figure A.5-9 LEU Flux Distribution for Energy Group 2 (0.10 – 1.0 MeV)



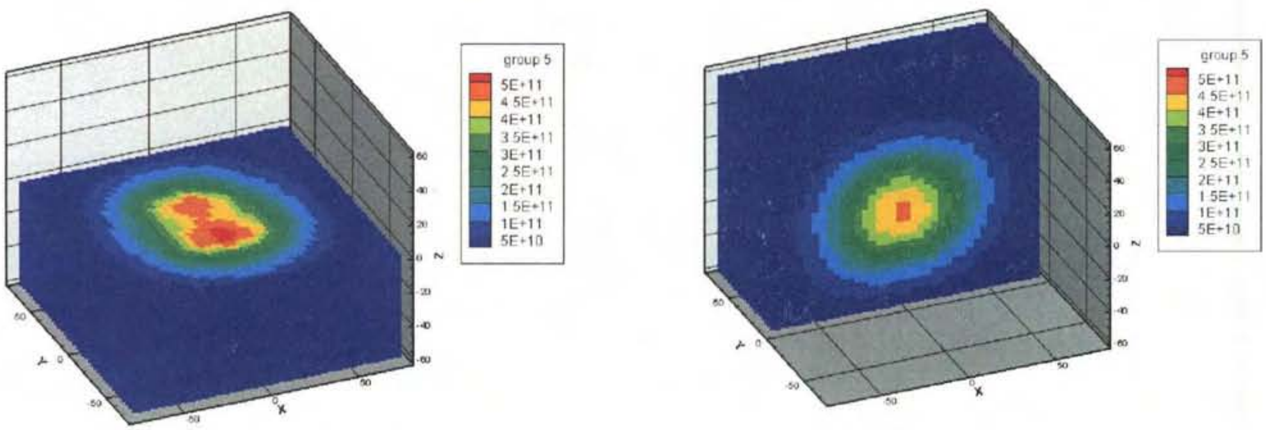
(a) x-y projection **(b) x-z projection**

Figure A.5-10 LEU Flux Distribution of Energy Group 3 (0.01 – 0.10 MeV)



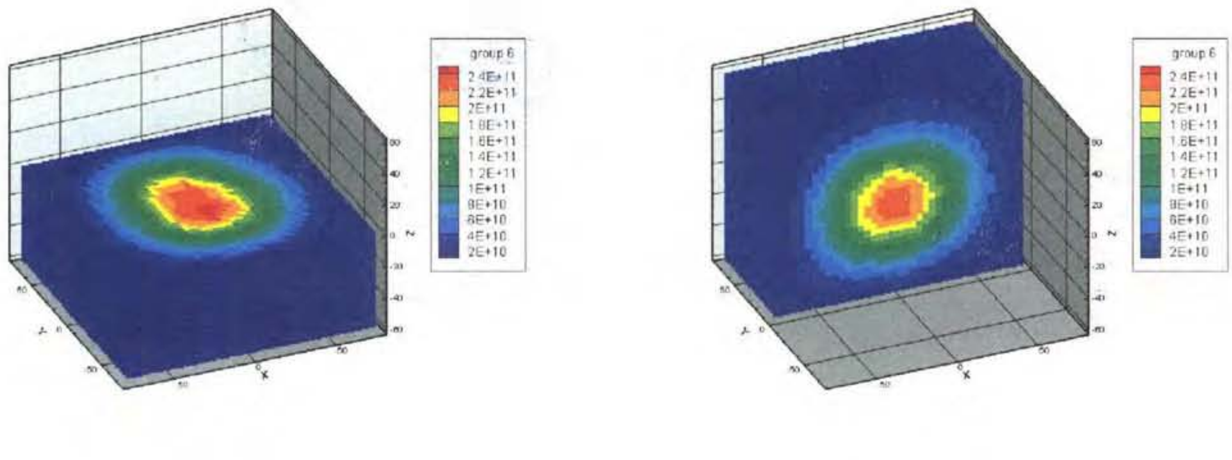
(a) x-y projection **(b) x-z projection**

Figure A.5-11 LEU Flux Distribution for Energy Group 4 (1.0 E-4 - .01 MeV)



(a) x-y projection **(b) x-z projection**

Figure A.5-12 LEU Flux Distribution for Energy Group 5 (1.0E-6 – 1.0E-4 MeV)



(a) x-y projection **(b) x-z projection**

Figure A.5-13 LEU Flux Distribution for Energy Group 6 (1.0E-7 – 1.0E-6 MeV)

A.6 Comparison of the parameters in the six-factor formula for HEU and LEU cores

The six-factor parameters provide an alternative way to characterize a reactor and provide insight into different physical processes occurring in the reactor core. Each factor represents a step in the "life cycle" of a neutron and is defined by specific parameters of the system, i.e. by compositions, cross sections and other nuclear properties. The six-factor formula is given by

$$k = \eta f \varepsilon p P_{FNL} P_{TNL} \qquad \text{(A.6.1)}$$

The first of factor (η) is the number of fission neutrons produced per absorption in the fuel. The second factor is the *thermal utilization* (f); it represents the effectiveness of the fuel in competing with other materials in the reactor for the absorption of the thermal neutrons. Then to account for the process of slowing down, two other factors are introduced. The third factor, the *fast fission factor* (ε), take into account that some of the fissions are produce by fast neutrons. The fourth factor, the *resonance escape probability* (p), represents the fraction of neutrons that managed to slow-down to the thermal energies without being absorbed. The last two factors are related to the probability of non-leakage and can be broken down P_{FNL} (*fast non-leakage*) and P_{TNL} (*thermal non-leakage*).

Table A.6-1 compares the calculated parameters of the six-factor formula for the HEU and LEU cores.

Table A.6-1 Six Factors for the Depleted HEU Core and the Reference LEU Core

Factor	HEU Value	LEU Value	Relative Diff. (%)
η	1.987	1.889	-4.932%
Thermal utilization f	0.613	0.640	4.405%
Fast fission factor ε	1.057	1.067	2.156%
Resonance escape probability p	0.945	0.937	-0.847%
Probability of non-leakage P_{NL} [1]	0.822	0.828	-0.730%

[1] Note that the fast and thermal non-leakage have been combine in one probability of non-leakage

A lower η value and a higher ε value are expected for the LEU core because of the low enrichment of U-235.

A.7 Results for thermal hydraulic analysis of LEU Core

Due to various uncertainties in the models, the as-built core can differ slightly from the proposed and analyzed core. Therefore, we have analyzed the impact of the number of plates in the partial fuel bundle 3-2 (see conversion SAR Figure 4-15) of the LEU core.

We analyzed two cases to evaluate the impact of the number of plate on the bundle peak-to-average power ratio, the power per plate in the critical bundle, and the width profile in the critical plate: i) A partial fuel bundle with 6 fuel plates, and ii) A partial fuel bundle with 13 fuel plates. Table A.7-1 compares the peak-to-average power ratio from the reference core (partial fuel bundle with 10 plates) with the two aforementioned cases.

Table A.7-1 Bundle Peak-to-average Power Ratio

Bundle #	*6 plates*	*10 plates*	*13 plates*	*Bundle #*	*6 plates*	*10 plates*	*13 plates*
1-1	0.945	0.933	0.924	4-1	1.004	0.968	0.943
1-2	1.052	1.046	1.039	4-2	1.128	1.085	1.059
1-3	1.040	1.026	1.014	4-3	0.902	0.865	0.842
1-4	1.175	1.164	1.154	4-4	0.996	0.955	0.931
2-1	1.104	1.111	1.110	5-1	1.173	1.129	1.106
2-2	1.075	1.101	1.110	5-2	1.111	1.075	1.055
2-3	**1.237**	**1.238**	**1.234**	5-3	1.034	0.990	0.968
2-4	1.206	1.228	1.232	5-4	0.985	0.945	0.924
3-1	0.933	0.973	0.996	6-1	0.917	0.896	0.878
3-2	0.412	0.658	0.836	6-2	0.831	0.812	0.800
3-3	1.033	1.075	1.101	6-3	0.825	0.797	0.781
3-4	0.881	0.930	0.963				

The **boldface** data highlights the critical bundle peak-to-average ratios.

In the above table, it can be observed that location of the critical bundle is not affected by the changes in the number plate. Moreover, the analysis showed that the maximum change in the critical power is about -0.5% for the case with 6 fuel plates. It is interesting to note that adding three additional fuel plates to bundle 3-2 has an insignificant impact on the power of the critical bundle. Table A.7-2 gives the plate peak-to-average power ratio in the critical bundle 2-3.

Table A.7-2 Plate Peak-to-Average power Ratio in the Critical Bundle

Plate #	Case: 6 plates	Case: 10 plates	Case: 13 plates
1	1.010	1.010	1.011
2	0.972	0.973	0.973
3	0.952	0.953	0.953
4	0.941	0.944	0.941
5	0.934	0.937	0.937
6	0.936	0.938	0.936
7	0.940	0.941	0.938
8	0.949	0.946	0.951
9	0.960	0.961	0.962
10	0.983	0.983	0.983
11	1.013	1.011	1.012
12	1.057	1.056	1.055
13	1.123	1.121	1.121
14	1.231	1.225	1.226

It can be seen in Table A.7-2 that the plate power distribution in bundle 2-3 is only slightly affected by the plate composition of the partial fuel bundle 3-2. However, this does not affect the selection of the critical plate.

The changes in the peak-to-average ratio power width profiles for the critical plate was also studied as a function of the number of plates in the partial fuel bundle. Figure A.7-1 shows the relative difference in the peak-to-average ratio power width profiles between the nomina; case (10 plates) and the two cases.

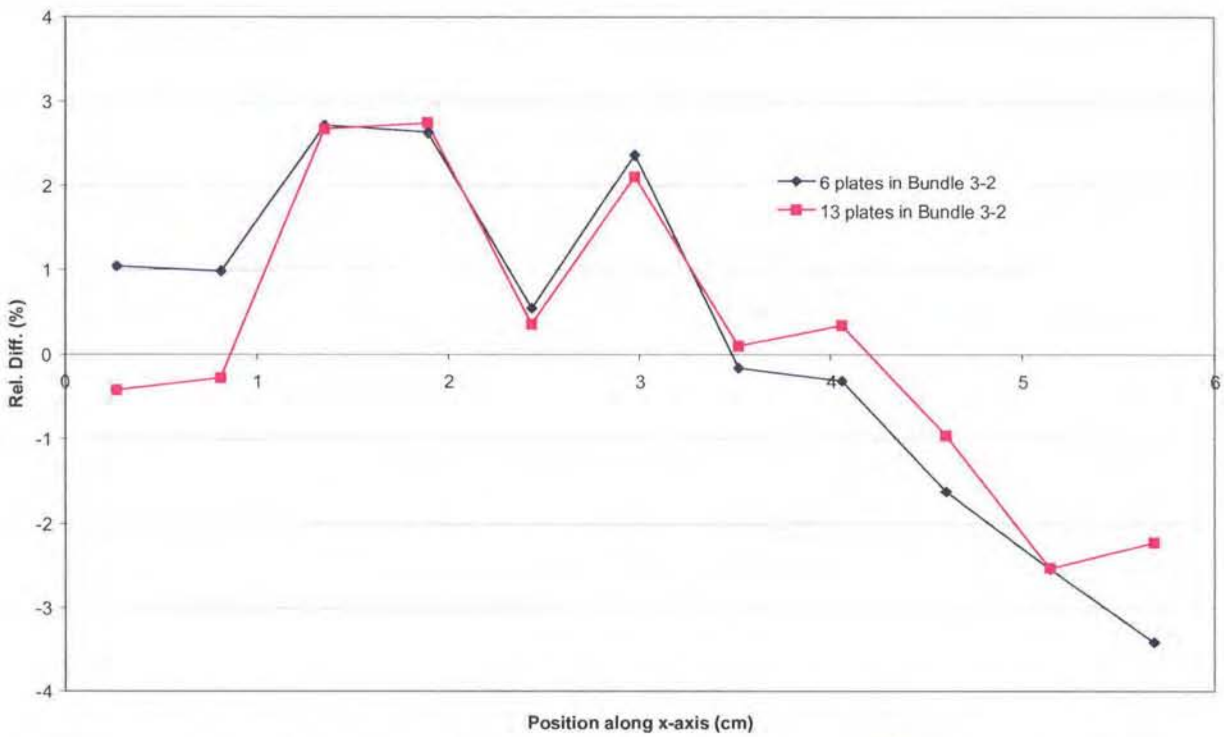


Figure A.7-1 Relative Difference between the Peak-to-Average Power Width Profile

The small changes in the power distribution presented in Table A.7-1, Table A.7-2 and Figure A.7-1 will have a minimal impact on the thermal-hydraulics analyses performed in the conversion submittal for the UFTR.

A.8 Additional Thermal Hydraulic Analyses

The grid plate that supports the four fuel assemblies in each fuel box is included in the hydraulic analysis because it affects the distribution of flow and causes the velocity distribution in each fuel box to be more uniform. The hydraulic model in the PLTEMP code assumes that the hydraulic resistance for each vertical coolant path through the fuel box has two components, a form-, or k-, loss and a frictional loss. A single k-loss represents all of the form losses, including those for the grid plate. The frictional loss is represented by the product of the local friction factor and the length-to-hydraulic-diameter ratio. Figure A.8.1 shows the flow network for an LEU assembly that is typical of all 24 assemblies in the six fuel boxes. The same diagram with 10 internal channels instead of 13 would also represent a typical HEU bundle. There are four channel types to be considered: 1) The internal channels between fuel plates of the same bundle the, 2) The channel formed between an end fuel plate and the fuel box wall 3) The central channel formed between the facing fuel plates of adjacent assemblies, and 4) The side channel formed between two assemblies whose adjacent sets of fuel plates are side-by-side. The fourth type of channel was assumed to not be heated since all fuel plates are perpendicular to it. This affected only the viscosity of the channel in the hydraulic analysis. The center and side channels traverse the middle of the fuel box and are each shared by two adjacent assemblies. Therefore, half of the total flow area of each of these two channels belongs to each of the sharing bundles, although the hydraulic diameter of each is based on the full channel size.

Since the ends of the side edges of the fuel plates are open where they abut the side channel, in theory there can be some flow between the fueled channels and the side channel through the center of the fuel box. However, in general, this lateral flow is expected to be small since the local pressure is expected to be essentially uniform at each axial level. The higher vertical flow velocities in the bigger channels, which have the larger hydraulic diameters, tend to keep the axial pressure drops through each of the parallel paths equal and the pressures uniform at each axial level. When the pressure is uniform at each axial level there is no mechanism for redistribution of flow among adjacent open channels. Hence, the one-dimensional hydraulics model depicted by Figure A.8.1 is deemed to be a reasonable representation of the expected behavior of a symmetric quadrant of an LEU fuel box.

As Figure A.8.1 shows, the PLTEMP model for an LEU assembly would have 16 parallel flow paths. For each of these parallel paths or channels the pressure drop, ΔP, is given by:

$$\Delta P = (K + fL/D) \times \rho V^2/2 \qquad (A.8.1)$$

where K is the equivalent K-loss value (discussed below), f is the friction factor for smooth-walled channels, L is the channel length, D is the channel hydraulic diameter, ρ is the coolant density, and V is the average coolant velocity in the channel in the core region. For laminar flow the value of f is affected by the shape of the channel. A value of 96 divided by the Reynolds number was assumed.

The single value of K represents not only the form losses at the inlet and exit to the fuel plates, but also the hydraulic resistance due to the grid plate. As Figure A.8.2 shows, the grid plate was manufactured from a solid 5-5/16 inch by 4-15/16 inch metal plate that was ¾ of an inch thick. On the bottom of the plate nine ¼-inch wide rectangular grooves were milled into the plate at ½-inch intervals. These grooves are ½-inch deep and ¼-inch wide and are parallel to the longer edge of the plate. A similar set of 12 grooves were milled on the top of the plate, but are parallel to the shorter edge of the plate. Thus, the two sets of grooves are perpendicular. Four fuel bundles in a single fuel box rest on top of each grid plate. The fuel plates are aligned perpendicular to the top set of grooves. Thus, in the horizontal plane where the fuel plates contact the grid plate, which is at the entrance to the coolant channels formed between the fuel plates, the grid plate blocks about half of the channel flow area.

The pressure drop due to a form loss produced by a sudden change in area is proportional to the square of the velocity of the flow through the smaller of the two areas. The grid plate can be thought of as having three layers – grooved top and bottom layers and a middle layer where the two sets of perpendicular grooves intersect. The flow inside each assembly can be thought of passing through seven different zones as it travels from below the bottom of the grid plate to above the top of the fuel assembly. Each change in zone represents a change in flow area and a form loss. These zones from bottom to top are: 1) Below the grid plate, 2) The bottom layer of the grid plate, 3) The middle layer of the grid plate, 4) The top layer of the grid plate, 5) The plane where the fuel plates contact the grid plate, 6) The fuel plate region, and 7) The region above the fuel plates.

The equivalent K provided in equation A.8.1 represents the effect of all the form losses due to all of the changes in area along the path. It also must include additional K-losses needed to account for the changes in direction of the flow as it goes through the circuitous path through the grid plate. These turns also increase the hydraulic resistance and the pressure drop. Thus, K can be represented as:

$$K = \sum_i \left(K_i \left(\frac{A}{A_i} \right)^2 \right) \tag{A.8.2}$$

where K_i is the local k-loss value along the flow path, A is the flow area over the core (i.e., fuel plate) region, and A_i is the local flow area that corresponds to K_i. The product of the area ratio in equation A.8.2 and V in equation A.8.1 is V_i, which is the local velocity corresponding to flow area A_i.

The smallest flow area is typically the largest contributor to K since it has the largest area ratio. The smallest flow area is the one through the middle layer of the grid plate, zone 3. The total flow area in zone 3 of the grid plate (where the top and bottom grooves meet) is about 25% of the cross sectional area of the empty fuel box. The total flow area in the fueled region of the fuel box, zone 6 (fuel plate region), is about 75% of the total flow area of the empty fuel box. Thus, A/A_i for zone 3 is about 75%/25%, or 3. The square of this area ratio is about 9. A representative value of K_i for zone 3 is about 1. Thus, this component of K in equation A.8.2 is about 9. Although all of the other components of K should be much smaller than 9 since the square of their area ratios are much smaller, including them can only cause K to be larger. Thus, the value of 5 used in the PLTEMP model appears to be a conservative value.

Because it is informative to know the sensitivity of the flow in the internal coolant channel to the value of K selected, a separate model of the flow network of Figure A.8.1 was produced in which K was varied and the relative velocities in each of the channels were predicted. The four types of channels have a common pressure drop between the grid plate inlet and the fuel assembly outlets. In the model there is an unknown flow rate for each channel type and the common pressure drop is an additional unknown. There is a pressure drop versus flow relationship, equation A.8.1, for each channel and also the sum of the flows of all channels must be equal to the total flow. Thus, the number of unknowns equals the number of equations, as it should. This relatively simple mathematical model was represented on computer spreadsheet, where the governing equations were easily solved.

The average velocity was determined by dividing the total volumetric flow rate for the fuel box by the total fuel box flow area in the fuel region. The relative velocity for each channel type was determined by dividing the channel velocity over the fuel region by the average velocity. The most limiting cases in PLTEMP are those in which the coolant inlet temperature is the highest and the reactor flow rate is the lowest. Therefore, for the current analysis, the inlet temperature was assumed to be 43° C and the reactor flow rate was assumed to be 23 gpm. Values of relative velocity were obtained for all four channels types for the LEU core for values of K from zero to infinity, Table A.8.1. The internal channels of each assembly had relative velocities that are less than 1. The other three channel types generally had relative velocities greater than 1. As the common value of K is increased, the values of the four velocities become more uniform. All relative velocities approach 1 as K approaches infinity.

For the fueled region of all channels, the Reynolds number was observed to be less than 2000. Thus, the flow in this region is always laminar. A friction factor, f, of 96 divided by the Reynolds number was used in equation A.8.1 in all cases. The coolant temperature in the side channel was assumed to be that of the inlet. The coolant temperature in the other channels was assumed to have an average value of 70° C, which is approximately the average of the inlet and expected maximum outlet coolant values. These temperatures affect the coolant viscosity, which affects the Reynolds number and the friction factor.

Since in the PLTEMP analysis for the Onset of Nucleate Boiling (ONB), the internal coolant channels are the most limiting factor, their flow rates are of greatest concern. For this channel type, a reduction in K from 5 to 2 causes a 21% reduction in flow, i.e., (0.570 – 0.452)/0.570. This causes a 26% increase in channel bulk coolant temperature rise, i.e., 0.570/0.452 – 1. Similarly, an increase in K from 5 to 10 causes a 16% decrease in channel bulk coolant temperature rise. Thus, large variations of K about the conservative value of 5 used in the analysis have only a relatively moderate effect on thermal performance.

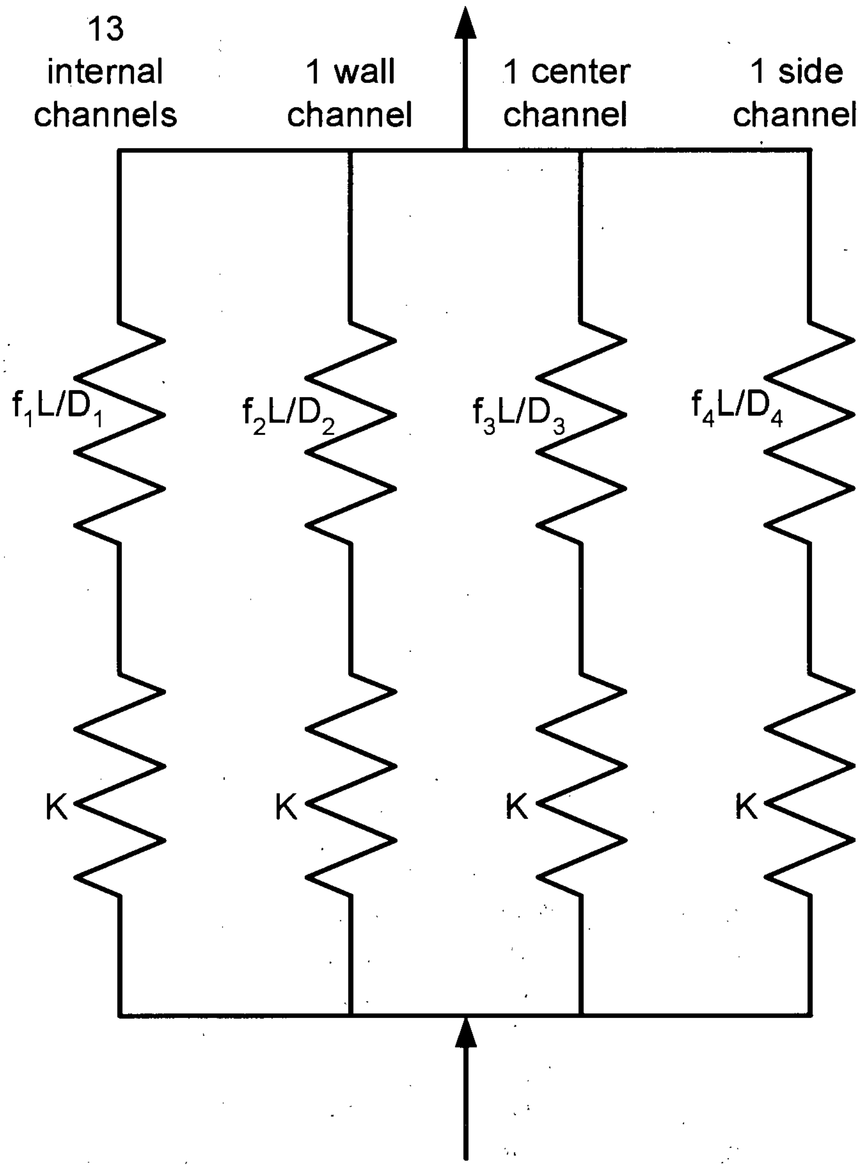


Figure A.8-1 Flow Network for One
LEU Assembly Including Grid Plate

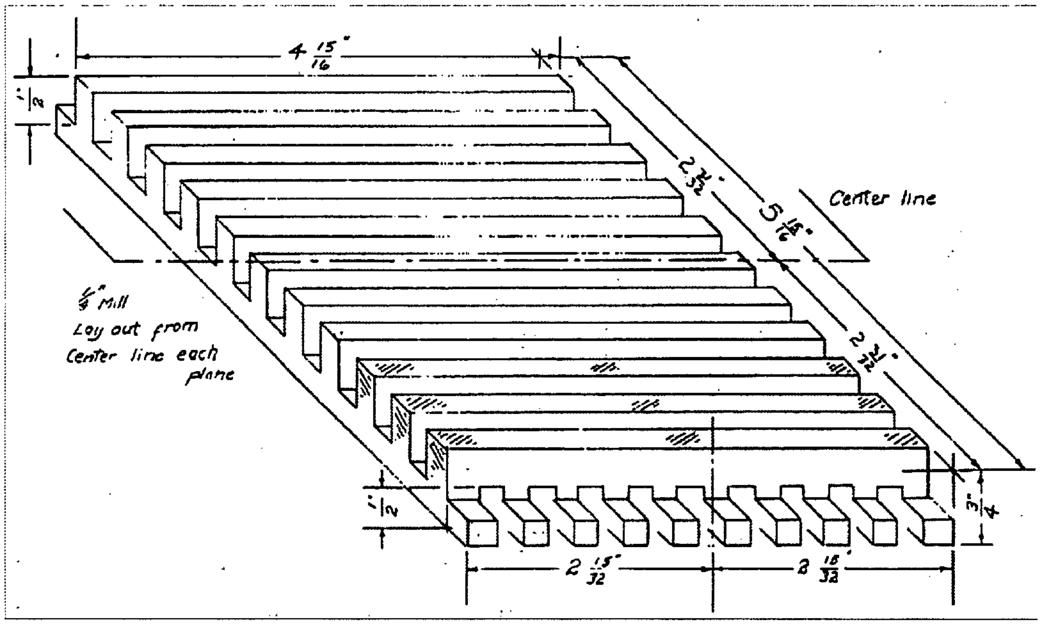


Figure A.8-2 Grid Plate

Table A.8-1 Relative Velocity

	Channel Type			
K	**Internal**	**Wall**	**Center**	**Side**
0	0.306	2.890	0.832	2.564
1	0.391	2.588	1.020	2.421
2	0.452	2.398	1.126	2.285
5	**0.570**	2.062	1.249	2.005
10	0.676	1.783	1.272	1.751
20	0.778	1.526	1.231	1.509
50	0.884	1.272	1.140	1.265
100	0.935	1.152	1.083	1.149
200	0.965	1.081	1.046	1.079
infinite	1.000	1.000	1.000	1.000

A.9 Additional Calculations (MCNP, CFD, etc.)

Figure A.9-1 provides a top view of the limiting fuel channel for the LEU core and the power distribution along the width of the fuel plate. The power distribution was obtained from an MCNP calculation in which the fuel meat was divided into 11 vertical strips of equal width.[4] A representative test problem was developed so that a one-dimensional hand-calculation that imitates the PLTEMP solution for single fuel plate between two identical coolant channels could be directly compared with a detailed computational fluid dynamics, CFD, solution that was performed with the aid of the STAR-CD code. The limiting criterion is the Onset of Nucleate Boiling (ONB), which is governed by the peak clad temperature at the outer surface of the fuel plate.

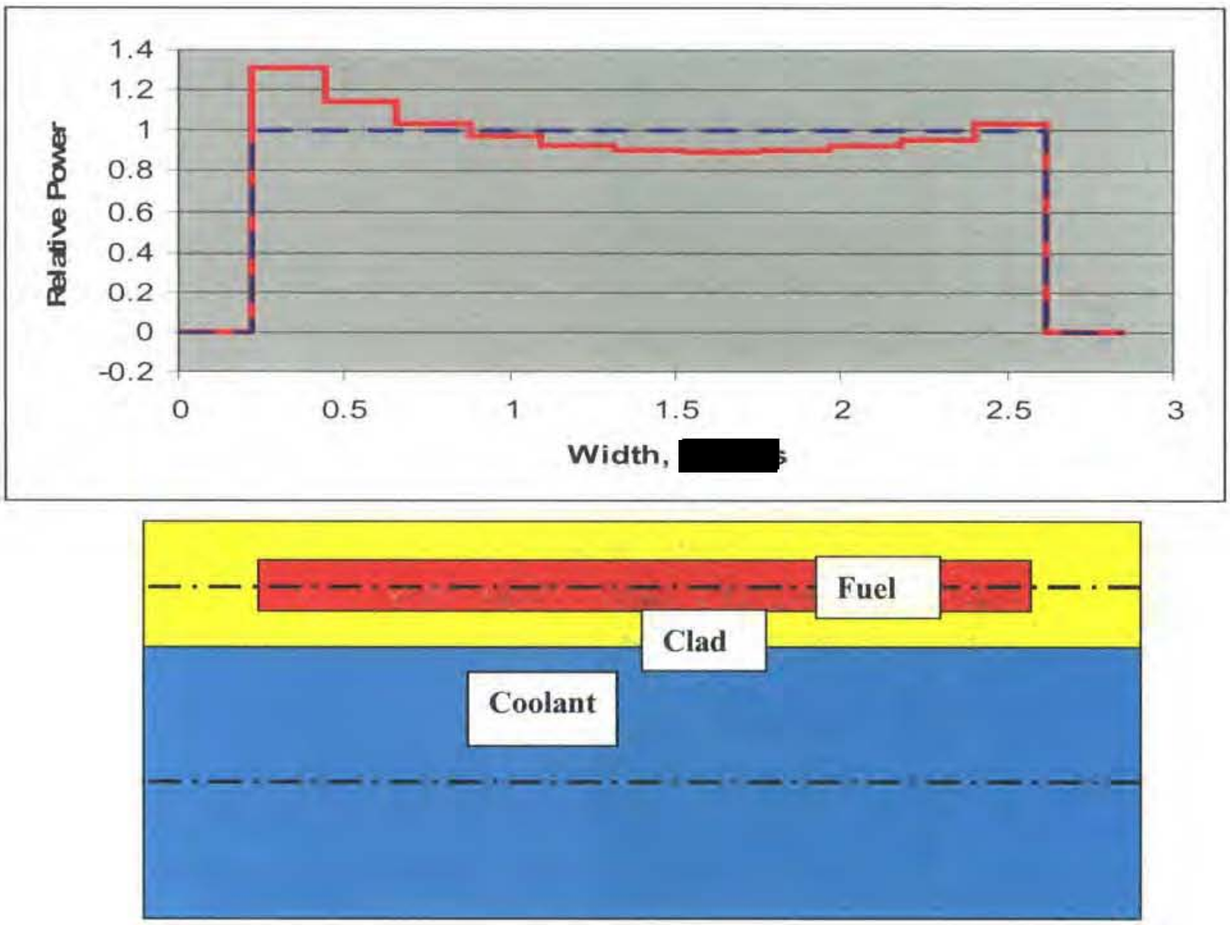


Figure A.9-1 Top View of Fuel Plate and Coolant Channel and Power Distribution along Hottest LEU Plate

[4] Subsequently, the MCNP calculation was redone causing a slight change in the power distribution from that shown in Figure A.9.1. The peak power increased from 1.314 to 1.325, for example. The differences have no impact on the analytical conclusions obtained in this appendix.

For the UFTR the peak clad temperature occurs at or near the top of the fuel meat. This is because the reactor has both an unusually low flow rate and an unusually low heat flux compared to the typical research reactor. The low flow causes the coolant temperature rise along the length of the fuel channels to be unusually high. The low heat flux causes the temperature rise from the coolant channel to the surface of the clad to be unusually low.

Figure A.9-2 shows the axial distribution of power along the length of the fuel plate for the limiting fuel plate. As the figure shows, the fuel was divided into 12 equal axial levels. For the representative test problem the plate power was assumed to be 350 W, the coolant inlet temperature was taken to be 30° C, and the coolant inlet velocity was assumed to be uniform and 8.25 mm/s. Two vertical planes, one through the center of the fuel meat and one through the center of the coolant channel, as shown in Figure A.9-1, were taken to be planes of symmetry. No-slip, i.e., zero-velocity, conditions were assumed to exist at all of the edges of the coolant channel shown in Figure A.9-1. The thin edges of the channel and the fuel plate were assumed to be insulated.

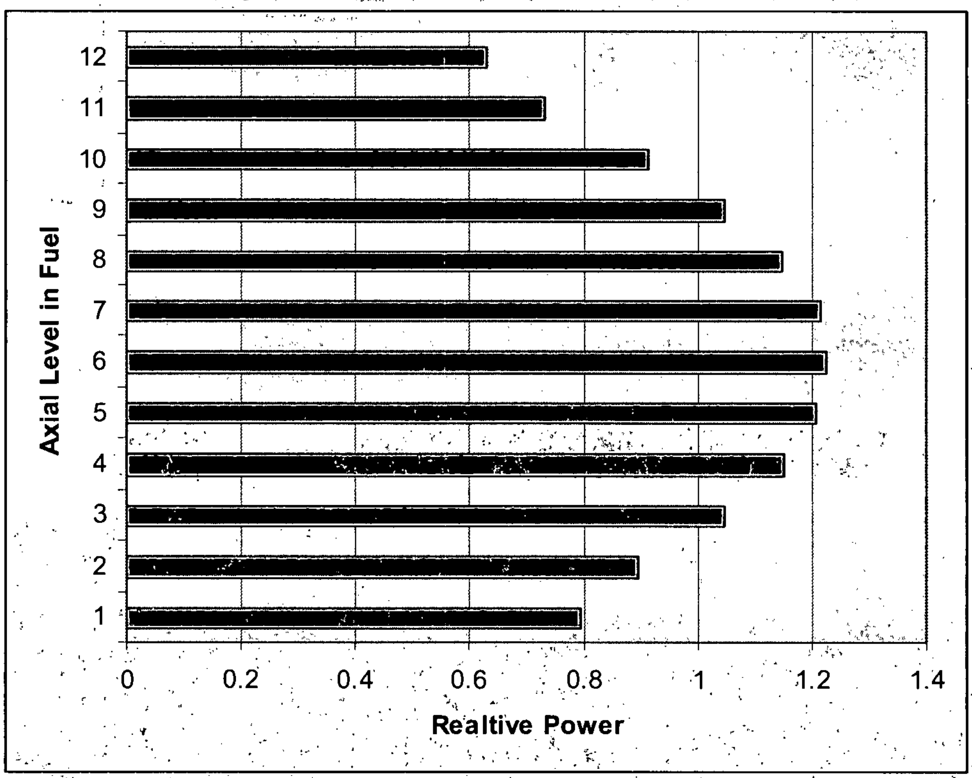


Figure A.9-2 Axial Power Distribution of Hottest LEU Fuel Plate

Since the peak clad temperature occurs very near to the top of the fuel meat, the one-dimensional solution for the peak clad temperature is easy to obtain. The test problem was designed so that the coolant temperature rise from inlet to outlet would be 50° C, which causes the coolant outlet temperature to be 80° C. The peak clad temperature is obtained by increasing this temperature by the film temperature rise from the coolant to the clad surface at the top of the fuel.

The film temperature rise is the local heat flux divided by the film coefficient. The average heat flux for the plate is obtained by dividing the 350-W power by twice the 23.625"-by-2.395"surface area of the face of the fuel meat. Thus, the average heat flux is 4794 W/m^2. As Figure A.9-2 shows, the relative power for the top level of the fuel plate is 0.6292. Therefore, the local heat flux is 4794 W/m^2 × 0.6292, or 3016 W/m^2. Since the flow is laminar, the Nusselt number – the film coefficient times the thermal conductivity of the coolant divided by the hydraulic diameter of the channel – is a constant value. For this situation, PLTEMP would use a Nusselt number of 7.63 for the nominal case and apply a hot channel factor of 1.2 as an uncertainty factor on the Nusselt number correlation. Hence, 7.63/1.2, or 6.36, would be used for the Nusselt number. Therefore, the film coefficient is 6.36 × 0.667 W/m-C / 5.43e-3 m, or 781 W/m^2-C. The film temperature rise at the top level of the fuel meat is 3016 W/m^2 divided by 781 W/m^2-C, or 3.9° C. Therefore, the peak clad temperature is 83.9° C.

For the CFD solution the region between the two vertical planes of symmetry shown in Figure A.9-1 and extending the entire length of the fuel plate was divided into about 200,000 computational cells. Figure A.9-3 shows the CFD solution for the surface temperature at the outer face of the clad. As the figure shows, the peak clad temperature is 83.3° C.

As a test of the CFD model, a second problem was performed in which the 350-W power was assumed to be uniformly distributed along the entire length and width of the fuel meat. For this case, the 0.6292 axial power factor is replaced by 1.0. This causes the film temperature rise to be 4794 W/m^2 divided by 781 W/m^2, or 6.1° C. Thus, the peak clad temperature for this case, based on the one-dimensional PLTEMP-type of model, is 86.1° C. Figure A.9-4 shows the CFD clad outer surface temperature distribution for this uniform power case.

As Figure A.9-4 shows, the peak clad outer surface temperature is 84.3° C. As expected, the temperature distribution is symmetric with the peak at the midpoint of the width. Table A.9-1 summarizes the peak clad temperature for both solutions to both problems. It is worth noting that the one-dimensional solution for the uniform power distribution predicts the peak clad temperature to be 1.9° C higher than is predicted by the CFD solution. A higher temperature is to be expected from the one-dimensional solution since it ignores heat conduction from the thin edges of the fuel meat to the adjacent clad. The nearly equal peak clad temperatures obtained from the two solutions for the two-dimension power distribution show that although the width-wise peak-to-average power ratio is greater than 1.3, there are substantial mitigating phenomena.

In conclusion, no special factors or adjustments to account for width-wise variations in fuel plate power distribution are needed to use the PLTEMP code to analyze the UFTR.

Table A.9-1 Summary of Results

Power distribution	Peak Clad Temperature (°C)	
	PLTEMP	CFD Solution
2-D	83.9	83.3
Uniform	86.2	84.3

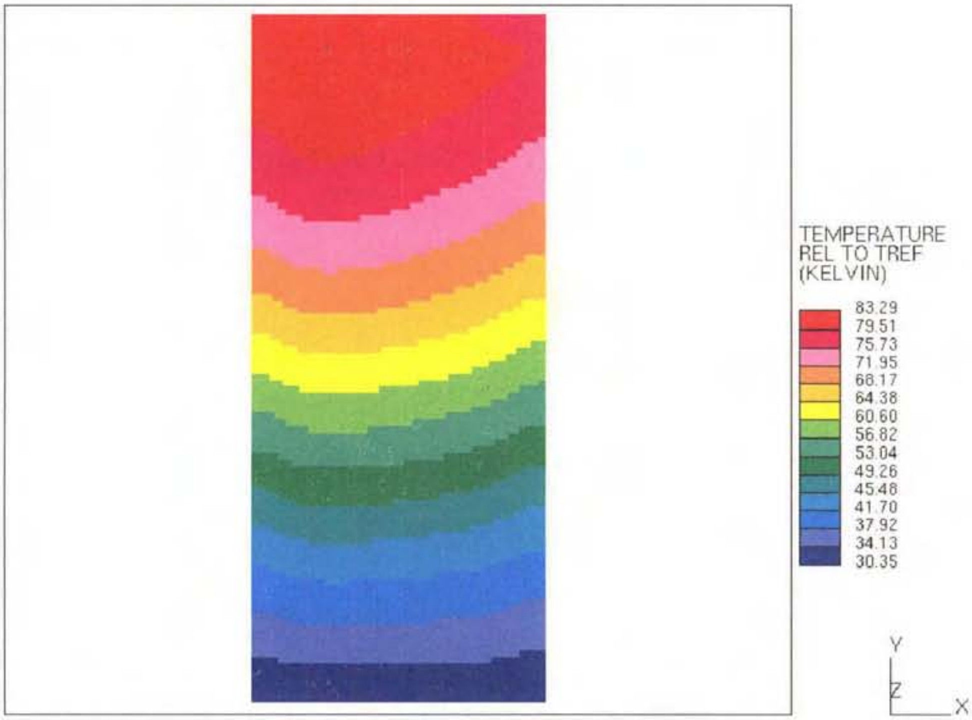


Figure A.9-3 Clad Surface Temperature Distribution Obtained from CFD Solution

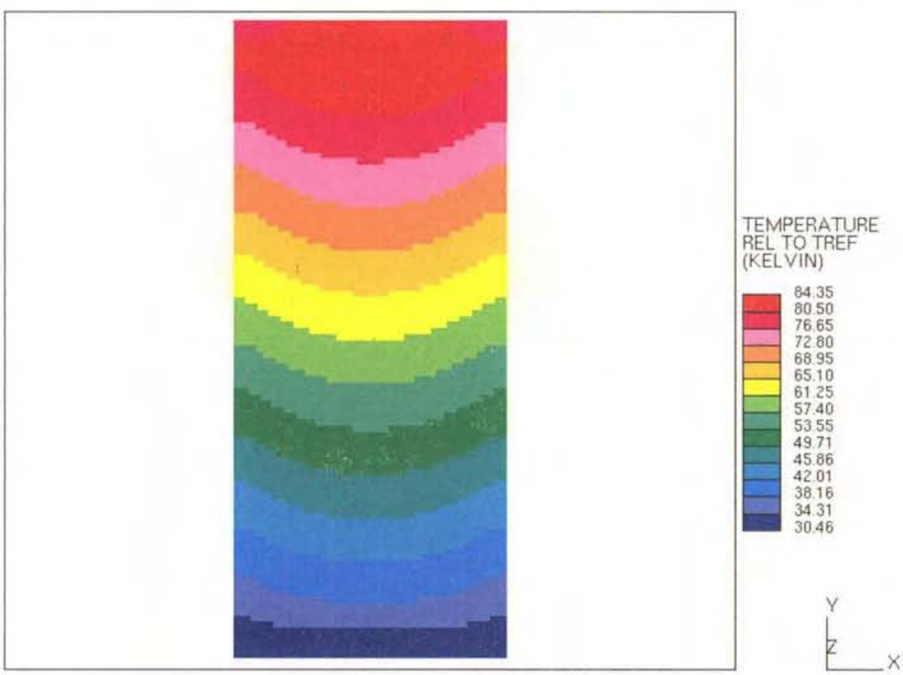


Figure A.9-4 Clad Surface Temperature Distribution Obtained from CFD Solution for Uniform Power Distribution

A.10 Methodology for determining the hot channel factors and for applying them in PLTEMP

I. Nominal Calculations

For the typical analysis performed for research reactors with the PLTEMP/ANL code, the most important quantity is the margin to the onset of nucleate boiling. If nucleate boiling is avoided then flow instabilities, which could rapidly lead to fuel failure, are avoided. The margins to flow instability and to critical heat flux are also evaluated. For research reactors the margin to nucleate boiling tends to be the most limiting criterion.

Nucleate boiling is assumed to occur when the temperature anywhere on the surface of any fuel plate reaches the temperature limit, T_{onb}. This limit is always greater than the local coolant saturation temperature, T_{sat}, by an amount ΔT_{sat}. ΔT_{sat} is a function of the local water pressure and the local value of heat flux on the surface of the fuel plate and is given by one of several available correlations and is typically several degrees Centigrade.

The local value of fuel plate surface temperature, T_{surf}, is given by:

$$T_{surf} = T_{in} + \Delta T_b + \Delta T_h \qquad (1)$$

Where, T_{in} is the inlet coolant temperature, ΔT_b is the bulk coolant temperature rise from the inlet of the reactor to the local plate elevation of concern, and ΔT_h is the local temperature rise from the bulk coolant to an immediately adjacent fuel plate surface.

ΔT_b and ΔT_h are given by:

$$\Delta T_b = \frac{q}{w\, c_p} \qquad (2)$$

and

$$\Delta T_h = \frac{q''}{h} \qquad (3)$$

where q is the power added to the coolant from the inlet to the elevation of interest, w is the flow rate in the channel, c_p is the specific heat capacity of the coolant, q'' is the local plate heat flux, and h is the local film coefficient at the surface of the fuel plate. Thus, PLTEMP calculates the fuel plate surface temperatures on all fuel plate surfaces at each axial level and compares each temperature to its allowed corresponding value of T_{onb}.

II. Limiting Calculations

A common approach in the analysis of nuclear reactors is to perform both a best-estimate calculation and a limiting calculation. For the former, all parameters, such as dimensions, power levels, flow rates, and heat transfer coefficients are set at their nominal, or best-estimate, values. A best-estimate analysis is a good first step in understanding the behavior of a system and assessing the feasibility of a design. It is also a gage against which limiting calculations can be judged. The limiting calculation includes the effects of manufacturing tolerances and operational and modeling uncertainties in the analysis.

A best-estimate calculation would employ nominal values in the evaluation of equation 1. For a limiting calculation hot channel factors F_{bulk} and F_{film} could be incorporated into equation 1, to produce:

$$T_{surf} = T_{in} + F_{bulk}\ \Delta T_b + F_{film}\ \Delta T_h \qquad (4)$$

where:

F_{bulk} is the uncertainty in bulk coolant temperature rise from reactor inlet to the local elevation of concern. (In the PLTEMP code documentation "F_{bulk}" is called "F_b".)

and

F_{film} is the uncertainty in the local film temperature rise at the location of concern on the fuel plate surface.

All hot channel factors are 1.0 for a best-estimate analysis and could be larger than 1.0 to include uncertainties in the limiting analysis. In the limiting calculation, nominal values of heat fluxes would be increased by a factor of F_q. Since ΔT_{sat} is a function of the heat flux, q'', increasing the heat flux by a factor of F_q also increases T_{onb}. Since ΔT_{sat} is typically only a several degrees, the effect may be small. Hot channel factors can also affect the other limiting criteria, such as the flow stability criteria.

III. Hot Channel Factors

Methods for determining hot channel factors for research reactors are described in References 1 and 2, which was intended for use in conjunction with earlier versions of the PLTEMP code. Some of these methods were employed in the construction of Table A.10-1. Two additional hot channel factors, not included in References 28 and 29, F_{film} and F_w, have been added. The former is in equation 4 and the latter is a divisor on flow/velocity and is to account for the variation in bulk coolant flow. F_w is not used in the analysis of the onset of nucleate boiling, but is used in some of the other limits that are evaluated by the PLTEMP code, such as those for flow instability.

Table A.10-1 Hot Channel Factors

						hot channel factors			
uncertainty	**type of tolerance**	**effect on bulk ΔT, fraction**	**value**	**toler-ance**	**toler-ance, fraction**	**heat flux, F_q**	**channel flow rate, F_w**	**channel temperature rise, F_{bulk}**	**film temperature rise, F_{film}**
fuel meat thickness (local)	random				0.00	1.00			1.00
U235 homogeneity (local)	random				0.20	1.20			1.20
U235 loading per plate	random	0.50			0.03	1.03		1.015	1.03
power density	random	0.50			0.10	1.10		1.050	1.10
channel spacing, inches or mm	random	1.00	0.111	0.001	1.009		1.028	1.028	1.009
flow distribution	random	1.00			0.20		1.20	1.20	1.000
Random errors combined						**1.23**	1.20	**1.21**	**1.23**
power measurement	systematic	1.00			**0.00***	1.00		1.00	1.00
flow measurement	systematic	1.00			**0.00***		1.00	1.00	1.000
heat transfer coefficient	systematic				**0.20**				1.20
systematic errors combined						1.00	1.00	1.00	1.20
product of random & systematic						1.23	1.20	1.21	1.47

*These have been set to zero because true power and true flow are used in the PLTEMP calculation. The effects of these two systematic uncertainties are included in the interpretation of the results.

Table A.10-1 lists random and systematic sources of uncertainty separately. The random sources can affect any fuel plate or coolant channel. However, it is unlikely that all of the sources can adversely affect the limiting location(s) in the reactor core simultaneously. The first four random sources relate to the distribution of power. The final two random sources affect channel spacing and flow distribution. The three systematic sources affect all regions of the core essentially equally.

The first two random uncertainties, which are caused by variations in the fuel meat thickness and ^{235}U homogeneity, are labeled "local" in that they are assumed to be hot-spot effects that affect the heat flux in only a local area with only minor perturbations in bulk coolant temperature. In some reactor designs, these variations can affect considerably more than a small local area. Since these sources of uncertainty affect the distribution of fuel rather than the total amount of it, the bulk coolant outlet temperature is not affected by these sources. However, the relocation of fuel so that it is closer to the coolant inlet can result in higher bulk coolant temperatures at locations upstream of the outlet. Where this is a concern, subcomponents for F_{bulk} from these sources should be included. When fuel meat thickness or the ^{235}U homogeneity subcomponents are included in F_{bulk}, it may not be appropriate to also include the ^{235}U loading per plate subcomponent in F_{bulk}.

The first four random uncertainties are assumed to affect only one of two plates that bound a coolant channel. Therefore, the effect on bulk coolant temperature rise, as represented by the corresponding F_{bulk} component, is assumed to be half as large. For example, a 3% fuel overloading in a single plate would produce a 1.030 F_q subcomponent, but only a 1.015 F_{bulk} subcomponent.

The systematic errors can be directly included in the PLTEMP calculation by increasing the reactor power, decreasing the reactor flow and decreasing the Nusselt number, which provides the film coefficient, to reflect the systematic errors. Then only the combined random errors need be modeled as direct multiplicative factors applied to calculated temperature rises and heat fluxes. This is what was done. Thus, the systematic errors are directly incorporated into the physics of the problem and the random errors are largely incorporated via equation 4. Although the product of the random and systematic errors provided in the bottom row of Table A.10-1 represent the total combination of hot channel factors, they are not used in the PLTEMP code.

A line-by-line description of Table A.10-1 follows:

<u>Fuel meat thickness (local)</u>
This is a result of the manufacturing process. When the fuel plates are rolled to the desired size, the fuel meat thickness in some regions of the plate may be thicker by as much as a specified tolerance. Other regions of the fuel meat can be too thin and result in less than the nominal heat flux. The amount of ^{235}U in each plate is assumed to be measured separately so that the fuel meat thickness only affects the distribution of power within the plate. The manufacturing specification for the LEU fuel is on the amount of fuel loading per unit area of the fuel plate. This specification causes the effect of fuel meat thickness variations to appear as part of the ^{235}U homogeneity specification since effects of both are measured in a single measurement.

Therefore, for the LEU fuel the fuel meat thickness tolerance is set to zero in Table A.10-1 and the ^{235}U homogeneity (local) is specified to include both effects together.

^{235}U homogeneity (local)

This is a tolerance on how well the ^{235}U is mixed with the other ingredients that are in the fuel meat. The amount of ^{235}U in each plate is assumed to be measured separately so that the ^{235}U homogeneity only affects the distribution of power within the plate. The 20% uncertainty shown in the table is considered to be typical for LEU fuel and to include the effects of variations in fuel meat thickness. For HEU fuel 3% is considered to be typical.

^{235}U loading per plate

This is a tolerance on the weight of ^{235}U that is to go into a plate. This must be based the manufacturer's specification for the fuel.

Power density

This uncertainty is assumed to be a result of the physics calculations and can result in more power being in a particular plate than was predicted and used in the nominal thermal-hydraulic analysis.

Channel spacing, inches

This tolerance is obtained by dividing the nominal channel thickness by the minimum channel thickness allowed by the dimensional tolerances. In Table A.10-1 1.009 was obtained by dividing 0.111 inches by (0.111 – 0.001) inches. For plate geometry where the hydraulic diameter can be approximated as twice the channel thickness, the formulas for obtaining the F_{bulk} and F_h subcomponents can be found on page 5 in Reference 29. They are as follows:

$$F_{bulk} = \left(\frac{t_{nc}}{t_{hc}}\right)^{\frac{3}{2-\alpha}} \quad (6)$$

$$F_h = \left(\frac{t_{nc}}{t_{hc}}\right)^{\frac{0.4+\alpha}{2-\alpha}} \quad (7)$$

where t_{nc} and t_{hc} are the nominal channel thickness and the minimum (or hot) channel thickness, respectively. α is the value of the Reynolds number exponent in the friction factor relationship. In this relationship, friction factor, f, is approximated as being proportional to $Re^{-\alpha}$. For turbulent flow α is typically 0.2 or 0.25. 0.25 was used in Table A.10-1. For laminar flow α is 1. Thus, for laminar flow, equation 6 reduces to the following:

$$F_{bulk} = \left(\frac{t_{nc}}{t_{hc}}\right)^{3} \quad (8)$$

This result is to be expected because when the flow is laminar, for a fixed pressure drop, the flow rate between two parallel plates is proportional to the cube of the channel spacing.

Equation 7 is based on the assumption that the flow is turbulent, which it the typical situation. When the flow is laminar, as it is for the UFTR, the Nusselt number is independent of flow rate and is a constant value. The heat transfer coefficient, h, is inversely proportional to hydraulic diameter, which is essentially equal to twice the channel thickness in plate reactors. Thus, for laminar flow, thinning the channel *increases* h. This presents a problem because thinning the channel also reduces the flow. Thus, for laminar flow, changing the channel thickness creates two opposing effects. For laminar flow, equation 7 should be replaced by:

$$F_h = \left(\frac{t_{hc}}{t_{nc}}\right) \quad (9)$$

Here the hot channel thickness, which is in the numerator, is that of the largest channel thickness allowed by the manufacturing tolerances. Obviously, the same channel cannot be both at the thinnest allowed by the manufacturing tolerances (equation 8) and at the same time also be at the thickest allowed by the manufacturing tolerance (equation 9). Employing such an assumption in the analysis, as was done for the UFTR, would be conservative and could be used to avoid having to consider both extreme thicknesses and all thicknesses in between. For the UFTR, ΔT_h, is only several degrees Celsius. There, the amount of conservatism is also small. For both laminar and turbulent flow the F_w subcomponent is equal to the F_{bulk} one.

Flow distribution

This uncertainty is the result of the hydraulic analysis that is used to determine the distribution of flow through the reactor. This is a local effect that does not systematically affect all coolant channels. Quantities, such as friction factors and form losses, and the influence of grid plates and fuel assembly side walls can not be precisely predicted. Although hydraulic models often predict that channels of equal thickness have the same channel average velocity, in plate assemblies of some research reactors the average velocities in the end coolant channels have been observed to be several percent less than that the average velocity of all of the coolant channels in the assembly. The 20% uncertainty assumed for the UFTR is based on engineering judgment.

Random errors combined

As suggested in the References 28 and 29, treatment of hot channel factors, it is unlikely that all of the random errors and uncertainties will occur together at the most limiting location in the reactor and that each will adversely effect reactor performance. Therefore, the random subcomponents, F^i, of each hot channel factor, F are combined statistically, i.e.,

$$F = 1 + \sqrt{\sum_i \left(1 - F^i\right)^2} .$$

Power measurement

This is a tolerance of the meter that is used to measure power and, if present, would affect all fuel plates essentially equally. Since true power is assumed in the PLTEMP calculations for the safety limit case, the tolerance is set to zero. However, systematic uncertainties are included in the interpretation of the results.

Flow measurement
This is a tolerance of the meter that is used to measure flow and, if present, would affect the flow in all flow channels essentially equally. Since true flow is assumed in the PLTEMP calculations the safety limit case, the tolerance is set to zero. However, systematic uncertainties are included in the interpretation of the results.

Heat transfer coefficient
This is due to uncertainties in the correlations for Nusselt number that are used to determine values of heat transfer coefficient, h. If the Nusselt number correlations that are used in the analysis predict values that are too large, then the predicted temperatures on all clad surfaces will be lower than would otherwise be experienced by the reactor. This is a core-wide effect rather than one that is random in location. A factor of 1.20, based on engineering judgment was used here.

Systematic errors combined
Because systematic errors, such as an error in reactor power and flow measurement, affect all locations within the reactor at the same time, it is reasonable to expect that all of them could be present at the limiting location(s). Therefore, the systematic subcomponents are combined multiplicatively, i.e., $F = \prod_i F^i$.

Product of random and systematic parts
Each of these products provides a hot channel factor, which represents the combination of all of its random and systematic subcomponents. However, these values are not directly used in the the PLTEMP code.

Table A.10-1 shows the results of two extreme methods of combining hot channel factors, a very conservative method that treats all contributors as if they were systematic and combines them multiplicatively and the opposite extreme, which is totally unacceptable and treats all contributors as if they were random and combines them statistically. Although neither of these extreme sets of results is recommended, the comparison of them with the set at the bottom of Table A.10-1 is informative.

IV. Treatment of Hot Channel Factors in the PLTEMP Code

For the sake of transparency and simplicity the PLTEMP code has be revised to do three sets of calculations (in a single run of the code) and provide a set of results for each as described in the following three steps:

1. A nominal, or best-estimate, calculation
   This is done with all hot channel factors set to 1.0. If there are no systematic uncertainties, then step 2 would not be performed.
2. A calculation that incorporates only the *systematic* uncertainties in power, flow, and heat transfer coefficient

For the UFTR the nominal power is multiplied by 1.00, the nominal flow is divided by 1.00, and the nominal Nusselt numbers, which are used to evaluate h, is divided by 1.20. The method of solution would otherwise be identical to that in the step 1 nominal, or best-estimate, calculation.

3. A final calculation that adds the effects of the *random* uncertainties to the solution obtained in step 2

   When step 2 is performed, sufficient information is stored for each location modeled in the core so that equation 4 can be evaluated at each location. The heat flux at each location on the fuel plate surfaces is also stored. Since the results of step 2 already include the higher power (not higher since true power used for UFTR), reduced flow (not reduced since true flow used for UFTR), and reduced heat transfer coefficient caused by the systematic errors, only the hot channel factors due to random errors are used here. These are also used in the correlations for the limiting criteria. The hot channel factor values shown in bold for F_{bulk} and F_{film} are used in equation 4 and the value of F_q shown in bold is applied to all of the stored fuel plate heat fluxes.

The above proposed treatment of hot channel factors enables complete results with hot channel factors included to be provided for all locations within the reactor core in a single solution of the PLTEMP code.

A single PLTEMP solution provides limiting results, including the effects of hot channel factors, for all locations represented by the PLTEMP model. Thus the process and the calculations that were used to determine the limiting safety case with hot channel factors included have been explained.

A.11 Detailed Discussion of Reactor Reload and Startup Plan

UFTR SOP-C.2 (Fuel Loading) will be used to control loading LEU fuel (U_3Si_2-Al) into the UFTR. A copy is available as updated accounting for changes required by use of LEU fuel. Additional guidance for following SOP-C.2 (Fuel Loading) is provided here. These additional instructions do not modify SOP-C.2 in any way; they only clarify it. The attached Figure A.11-1 and Table A.11-1 provide a working record of new fuel bundles and the order and location for their loading in the reactor core. The following summarizes the loading sequence:

- All fuel loading shall be made from the most reactive to the least reactive locations.

- Fuel will be added in seven fuel load increments:

    1. Load NC and SC fuel boxes (insert 8 fuel bundle numbers).

    When the NC and SC fuel boxes are loaded, insert wedge pins to assure proper fit.

    2. Load one fuel assembly each in NW and SE fuel boxes (insert 2 numbers).
    3. Load one fuel assembly each in NE and SW fuel boxes (insert 2 numbers).
    4. Load one fuel assembly each in NW and SE fuel boxes (insert 2 numbers).
    5. Load one fuel assembly each in NE and SW fuel boxes (insert 2 numbers).
    6. Load one fuel assembly each in NW and SE fuel boxes (insert 2 numbers).
    7. Load one fuel assembly each in NE and SW fuel boxes (insert 2 numbers).

Subsequent bundles and partial bundles will be loaded individually; the remaining dummy fuel bundles and wedge pins will be installed last. Since the approach-to-critical is an experimental procedure with uncertainties, the exact final core loading necessary for ~1% $\Delta k/k$ is not specified here. Table A.11-1 is a listing of the core fuel loading including fuel bundle serial numbers, gm U/gm U-235 content and core location, while Figure A.11-1 is a map of the core to show LEU fuel bundle locations. Both tables are incomplete until fuel bundle information is obtained upon loading. Since all fuel bundles are nominally identical (U/U-235 loading) this is not a limitation, but records will be maintained of the as-loaded core. As set up, the table assumes initially that the LEU loading will be the calculated 22 full bundles with a 10-plate partial fuel bundle in the southeast fuel box plus a partial dummy bundle in the southeast box and a full dummy bundle in the northeast box. Should less fuel be required (less likely), then dummy bundles will replace the last or even the penultimate fuel bundle loaded. If more fuel is needed, a full fuel bundle will replace the partial bundle in the southeast fuel box with a partial-up-to-full fuel bundle then loaded into the northeast fuel box. Cores containing 21 fuel bundles up to 24 fuel bundles are considered acceptable.

If the reactor is predicted to become critical during Increments 6 or 7, no more than one full or partial fuel assembly shall be added to the core in any one step during all subsequent loading to the prescribed excess reactivity loading. The intent is to load ~1.0% $\Delta k/k$ excess reactivity since this will provide sufficient maneuverability for the near term. All approach-to-critical (inverse multiplication) plots will be retained as part of loading records and the critical loading will be checked versus calculations.

After reception, satisfactory inspection/checkout, and loading of the new U_3Si_2-Al fuel following UFTR SOP-C.2 (Fuel Loading), a number of surveillances and activities will need to be completed in addition to assuring practical training requirements are maintained for licensed SROs. The plan for completion of the remaining surveillances and activities will be as listed in Table A.11-2. Items 3 and 4 are repeated as Items 8 and 9 to assure restacking shielding has not affected control blade movement prior to continuing. These surveillances and activities are ordered in a conservative manner; that is, surveillances and activities are completed at lower power levels prior to operation at higher power levels to assure protection of the health and safety of the facility and staff and public. The scram checks surveillance is performed first as the most important at shutdown.

Both SRO W. G. Vernetson and SRO M. A. Berglund will have maintained their license status during the HEU to LEU fuel conversion outage by performing at least 4 hours of SRO activities as SROs per quarter during the outage. However, both SROs will be past the limit listed in the Requalification and Recertification Training Program for quarterly performance of a reactor startup and shutdown. Therefore, both will participate in the restart plan as part of their training. Specifically, they will complete the first weekly checkout with LEU fuel together (all steps performed or observed by each SRO), and each will perform a complete daily checkout observed by the other. Subsequently, after performance of all the non-operational surveillances, each will perform a 1-Watt startup to verify proper instrumentation response and expected critical position.

Upon successful completion of the surveillances and activities listed in Table A.11-2, the UFTR will be returned to normal operation. Records of all these surveillances along with complete documentation for the new core will be maintained together for ease of examination and audit.

Table A.11-1 UFTR LEU Core Fuel Loading form 6M Drums

Table 1 tracks the fuel bundle movements to be made to load the LEU fuel into the UFTR core.

Movement Number	Fuel Bundle Serial Number	Gm U/Gm U-235[1]	Core Location
1		/	SW of NC
2		/	SE of NC
3		/	NW of SC
4		/	NE of SC
5		/	NW of NC
6		/	NE of NC *(WP)*[2]
7		/	SW of SC
8		/	SE of SC *(WP)*[2]
9		/	SE of NW
10		/	NW of SE
11		/	SW of NE
12		/	NE of SW
13		/	NE of NW
14		/	SW of SE
15		/	NW of NE
16		/	SE of SW
17		/	SW of NW
18		/	NE of SE
19		/	NW of SW
20		/	SE of NE
21		/	NW of NW
22		/	SW of SW
23		/	SW of SE (partial fuel)
24		/	SW of SE (partial dummy)
25		/	NE of NE (full dummy)
TOTAL:[3] 25 Movements; ________Gm U /________Gm U-235			

[1] These values represent initial fuel loading quoted from receipt records.
[2] *WP* implies wedge pin insertion.
[3] This is based upon calculations expected to load 22 full fuel bundles with a 10-plate partial fuel bundle and a partial dummy bundle in the SE fuel box and a full dummy bundle in the NE box. If more fuel is needed or fuel loading is terminated early, then Table A.11-1 will be used to record actual loading.

UFTR LEU CORE FUEL LOADING FROM 6M DRUMS (Date: __________)

21 NW of NW	13 NE of NW	5 NW of NC	6 NE of NC	15 NW of NE	24D NE of NE
17 SW of NW	9 SE of NW	1 SW of NC	2 SE of NC	11 SW of NE	20 SE of NE

19 NW of SW	12 NE of SW	3 NW of SC	4 NE of SC	10 NW of SE	18 NE of SE
22 SW of SW	16 SE of SW	7 SW of SC	8 SE of SC	14 SW of SE	23 23D SW of SE

Figure A.11-1 Template of Fuel Box Contents Noting Fuel Loading Movement Number

NORTH

Table A.11-2 Surveillance and Activity Completion Plan Following Loading of LEU Fuel

1. Successful completion of Preoperational and Radiation Protection Weekly Checkouts per SOP-A.1 and SOP-D.1. *[TS 4.2.2(6)(a) and TS 4.2.8(1)–(4)]*
2. Successful completion of Daily Checkout per SOP-A.1. *[TS 4.2.2(3) and TS 4.2.2(6)(b)]*
3. S-1 – Measurement of Control Blade Drop Times per SOP-0.5. *[TS 4.2.2(1)]*
4. S-5 – Measurement of Control Blade Controlled Insertion Times (and Withdrawal Times) per SOP-0.5. *[TS 4.2.2(2)]*
5. Restack all shielding for normal operations.
6. Successful completion of Daily Checkout per SOP-A.1. *[TS 4.2.2(3) and TS 4.2.2(6)(b)]*
7. Q-1 – Scram Checks (Safety System Operability Tests) per SOP-0.5. *[TS 4.1(2) referencing TS Table 3.2]*
8. S-11 – Replacement of Control Blade Clutch Current Light Bulbs per SOP-0.5. [*NRC Commitment]*
9. S-1 – Measurement of Control Blade Drop Times per SOP-0.5. *[TS 4.2.2(1)]*
10. S-5 – Measurement of Control Blade Controlled Insertion Times (and Withdrawal Times) per SOP-0.5. *[TS 4.2.2(2)]*
11. A-2 – UFTR Nuclear Instrumentation Calibration Check and Heat Balance (Pre-Calorimetric) per SOP-E.4. *[TS 4.2.2(8)(a)–(e)]*
12. 1 Watt Startup by SRO 1 (observed by SRO 2) to verify proper instrumentation response, and expected critical position per SOP-A.2.
13. 1 Watt Startup by SRO 2 (observed by SRO 1) to verify proper instrumentation response, and expected critical position per SOP-A.2.
14. B-1 – Verification of Negative Void Coefficient of Reactivity (1 Watt) per SOP-E.8. *[TS 4.2.1(3)]*

15. S-2 – Reactivity Measurements (Worth of Control Blades, Total Excess Reactivity, Reactivity Insertion Rate and Shutdown Margin) per SOP-A.7. All values checked versus predicted values. *[TS 4.2.1(1) referencing TS 3.1(1)–(4)]*

Table A.11-2 *contd.*

Surveillance and Activity Completion Plan Following Loading of LEU Fuel

16. A-2 – UFTR Nuclear Instrumentation Calibration Check and Heat Balance (First Power Run) per SOP-E.4. *[TS 4.2.2(8)(a)–(e)]*

    Start-up performed by SRO, with stops at:

    - *1 W* to verify proper instrumentation response, and expected critical position.
    - *100 W* to verify proper instrumentation response.
    - *1 kW* to verify proper instrumentation response, and conduct partial restricted area radiological survey to verify shielding effectiveness.
    - *10 kW* to verify proper instrumentation response, and conduct partial restricted area radiological survey to verify shielding effectiveness.
    - *90 kW* to verify proper instrumentation response, and conduct partial restricted area radiological survey to verify shielding effectiveness.

17. Q-4 – Radiological Survey of Unrestricted Areas (During First Full Power Run) per SOP-0.5. *[TS 3.9.2(3)(a)]*

18. Q-5 – Radiological Survey of Restricted Area (During First Full Power Run) per SOP-0.5. *[TS 3.9.2(3)(b)]*

19. S-4 – Measurement of Argon-41 Concentration (Conducted during Second Power Run for A-2 Surveillance per SOP-E.4, if possible, or in a Separate Run per SOP-E.6). *[TS 4.2.3(1) and TS 4.2.4(2)]*

20. A-3 – Measurement of UFTR Temperature Coefficient of Reactivity per SOP-E.7. *[TS 4.2.1(2)]*

21. Additional measurements to verify Neutron Flux Values in Experimental Ports (Center Vertical Port and Rabbit System to be within 20% of expected calculated value).

Appendix B

B.1 Engineering Uncertainty Factors

This attachment addresses the engineering uncertainty factors (or hot channel factors) that were used to compute the thermal-hydraulic safety limits, safety margins, and safety system trip settings in HEU and LEU cores. The rationale for choosing these factors and the method used to combine them are outlined along with a summary of results.

The PLTEMP code (Ref. 11) used in the analyses allows for introduction of three separate engineering hot channel factors as they apply to the uncertainty in the various parameters (as opposed to a single lumped factor). The three hot channel factors are:

F_q for uncertainties that influence the heat flux q
F_b for uncertainties in the temperature rise or enthalpy change in the coolant
F_h for uncertainties in the heat transfer coefficient h.

The code also allows introduction of nuclear peaking factors for the radial, F_r, and axial, F_z, distributions of the heat flux.

While there is no generally accepted method for the selection of hot channel factors, these factors are normally a composite of sub-factors, and the sub-factors can be combined either multiplicatively, statistically [$F_b = 1 + \sqrt{\sum(1 - fbi)2}$], or as a combination of the two. A detailed description of methods for calculating hot channel factors is contained in Ref. 27. The pure multiplicative method of combining the sub-factors is very conservative and somewhat unrealistic. The pure statistical method recognizes that all of these conditions do not occur at the same time and location and treats all of the uncertainties as random. The most realistic method is probably a combination of the multiplicative method for systematic uncertainties and the statistical method for the random uncertainties. The combined method is used for the analyses considered here.

The thermal and hydraulic design in Section 4.4 of the UFTR Final Safety Analysis Report dated January 1981 used a conservative method with "hot-channel factors" to calculate the fuel plate heat transfer data. In order to compare the HEU and LEU cores on a common basis using the methodology used in the PLTEMP code, the uncertainty factors shown in Table B.1-1 were identified for both the HEU and LEU cores. The systematic factors were combined multiplicatively and the random factors were combined statistically. The products of the two resulting factors were used for F_q, F_b, and F_h in calculations of the HEU and LEU cores.

Table B.1-1 HEU and LEU Engineering Uncertainty Factors

Uncertainty	Type	HEU F_q	HEU F_b	HEU F_h	LEU F_q	LEU F_b	LEU F_h
Fuel Meat Thickness (local) [a]	random	1.07	-	-	1.07	-	-
^{235}U Loading Per Plate [b]	random	1.03	1.015	-	1.03	1.03	-
^{235}U Homogeneity (local) [c]	random	1.03	-	-	1.20	-	-
Coolant Channel Spacing [d]	random	-	-	1.01	-	1.067	-
Power Level Measurement [e]	systematic	1.05	1.05	-	1.05	1.05	-
Calculated Power Density [e]	random	1.10	1.05	-	1.10	1.10	-
Coolant Flow Rate [e]	systematic	-	1.10	-	-	1.10	-
Heat Transfer Coefficient [e]	systematic	-	-	1.20	-	-	1.20
Pure Multiplicative Combination		1.31	1.23	1.21	1.53	1.40	1.20
Pure Statistical Combination		1.14	1.12	1.20	1.24	1.17	1.20
Random Factors Combined Multiplicatively		1.13	1.05	1.01	1.24	1.12	1.00
Systematic Factors Combined Statistically		1.05	1.16	1.20	1.05	1.16	1.20
Product of Random & Systematic Factors		**1.19**	**1.22**	**1.21**	**1.30**	**1.30**	**1.20**

[a] HEU: Estimated from HEU fuel plate thickness data.
LEU: Derived from fuel plate thickness specification of 50 ± 2 mils.

[b] HEU: Assumed to be the same as for the LEU plate.
LEU: Derived from fuel plate loading specification of [redacted]

[c] HEU: Estimated for U-Al alloy fuel meat.
LEU: From fuel plate homogeneity specification.

[d] HEU: Derived from tolerances on drawing UTR-103.
LEU: Derived from specification of 94 ± 2 mils for channel adjacent to fuel box and fuel plate thickness specification of 50 ± 2 mils.

[e] HEU and LEU: Assumed values.

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