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

SUBMITTAL OF RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION FOR THE CONVERSION FROM HIGH-ENRICHED URANIUM TO LOW-ENRICHED URANIUM FUEL

REDACTED VERSION

SECURITY RELATED INFORMATION REMOVED

Redacted text and figures blacked out or denoted by brakets

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

Amendment 26
(previously Amendment 25)
UFTR Technical Specifications

University of Florida Training Reactor, Facility License: R-56, Docket No. 50-83
Request for Change in Technical Specifications Approving HEU to LEU Conversion
<u>With Responses to Requests for Additional Information</u>

A proposed amendment to the UFTR Technical Specifications (R-56 License) for conversion from high enriched uranium (HEU) fuel to low enriched uranium (LEU) fuel affecting pages 4, 5, 6, 7, 8, 9, 13, 15, 16, 21, 23, 24, 26 and 38 of the approved Tech Specs is attached. The proposed changes will constitute Amendment 26 to the UFTR R-56 License as noted on the text pages. The changes are marked with the usual vertical line(s) in the right-hand margin with all amendments to date indicated on the bottom left on these Tech Spec pages.

This amendment is requested to allow conversion of the UFTR to operate with low enriched uranium fuel. Attachment I to this letter contains the actual changed pages for the Tech Specs. These changes are updated from the original submittal to obtain approval for HEU to LEU conversion dated December 2, 2005 that was referenced as Amendment 25. A separate Amendment 25 has since been approved; therefore, this submittal is now for Tech Spec Amendment 26. Attachment II is the additional safety analysis and responses to the Requests for Additional Information dated May 2, 2006 (28 questions) and May 22, 2006 (1 question) with all changes based on following the recommended format of NUREG-1537 (Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors) for such conversions.

For the UFTR HEU to LEU conversion, the only changes required for the UFTR Technical Specifications involve the fuel type and certain related specifications based upon the analyses performed.

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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 a single new specification on maximum clad and fuel temperature, correlated with the changed limiting safety system setting (LSSS) (trip points) on power level of 119 kW, flow rate of 36 gpm and primary coolant inlet temperature of 109° F and the accident analysis results presented to assure conservative limits in Section 4.7 of Attachment II as follows:

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

As noted in Attachment II to this and the earlier submittal, in the Section 4.7 analyses, the three parameters of power level, flow rate and primary coolant outlet temperature are interdependent. Basically, the fuel and cladding are adequately protected by appropriate settings of the limited safety system settings (LSSSs). The LSSSs that were selected are thus conservative settings that preempt 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 of Attachment II 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 986° F during steady-state operation. With the LSSSs 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, then the safety limit is assured.

Second, on page 5 of the Tech Specs, in Section 2.2, Limiting Safety System Settings, specifications (1), (2) and (3) become (1), (2) and (3)(a) and 3(b), respectively, as follows:

(1) 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 watt.
(3) The average primary coolant
    (a) inlet temperature shall not exceed 109° F
    (b) outlet temperature shall not exceed 155° F when measured at any fuel box outlet.

Specifications (1), (2) and (3)(a) restrict UFTR operations to the "Operating Region" of Figure 4-20 in Section 4.7.3.3 of the HEU to LEU safety analysis, referenced in Attachment II, taking into account uncertainties in measurement of reactor power, coolant flow rate and average primary coolant inlet temperature. Specification (3)(b) is returned to what specification (3) was prior to approval of Tech Spec Amendment 25 because the safety limit is based on clad temperature to protect the fuel so the LSSSs are maintained on all fuel box outlet temperatures at 155° F.

Third, on page 6 of the Tech Specs, in Section 3.1, Reactivity Limitations, item (2), the core excess reactivity at cold critical, without xenon poisoning, is changed from not exceeding 2.3% Δk/k to not exceeding 1.4% Δk/k, again based on the accident analysis results presented in Section 13 of Attachment II of the original December 2, 2005 submittal and considering the actual realistic excess reactivity needed for operations.

Fourth, also on page 6 of the Tech Specs, in Section 3.2.1, Reactor Control System, item (4), the limit on the control-blade-drop time is changed from not exceeding 1 sec to not exceeding 1.5 sec in agreement with the conversion analysis in Attachment II, Section 13.1 showing that raising the drop time has essentially no impact on temperatures reached in a reactivity transient.

Fifth, on page 7 of the Tech Specs, in Section 3.2.3, Reactor Control and Safety Systems Measuring Channels, the number of coolant temperature indicators required on the primary is changed from 6 to 7 to account for 1 inlet temperature plus 6 fuel box outlet temperatures per the LSSS changes.

Sixth, on page 8 of the Tech Specs, in Table 3.1, Specifications for Reactor Safety System Trips, the new high primary coolant average inlet temperature rod-drop trip at 109° F is added per the analysis in Attachment II.

Seventh, on page 9, in Table 3.2, Safety System Operability Tests, the test of the high average primary coolant inlet temperature is specifically added as a requirement of the daily checkout at the same frequency as for testing the outlet temperature per Attachment II.

Eighth, on page 13 of the Tech Specs, in Section 3.5, Limitations on Experiments, in paragraph (3)(a), the absolute reactivity worth of any single movable or nonsecured experiment shall not exceed 0.6% Δk/k is maintained but is now noted to include the combined worth of all such experiments as has always been understood but in agreement with the LEU conversion analysis in Attachment II looking at the 0.6% Δk/k transient reactivity insertion. In paragraph (3)(b), the limit on total absolute reactivity worth of all experiments is changed from not exceeding 2.3% Δk/k to not exceeding 1.4% Δk/k to be consistent with the change made on overall reactivity limitations per the allowed core excess reactivity noted in the third item above referencing page 6.

Ninth, on page 15 of the Tech Specs, in Section 3.7, Fuel and Fuel Handling, specification (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 of Attachment II of the original conversion submittal for the LEU fuel.

Tenth, on page 16 of the Tech Specs, in Section 3.8, Primary Water Quality, specification (5) is added to assure the water pH is in an acceptable range per the response to the RAI, Question 2.

Eleventh, on page 21 of the Tech Specs, in Section 4.2.7, Surveillance Pertaining to Fuel, based on the results of accident releases in Section 13.3 of the conversion analysis, Section 4.2.7(1) in Attachment II 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 to limit the possible/potential consequences of fuel handling accidents and to preclude damaging a fuel bundle with a dropped shield block before three days have elapsed.

Twelfth, also on page 21, in Section 4.2.8, Primary and Secondary Water Quality Surveillance, a new item (4) is added to require measurement of the primary water pH as part of the weekly preoperational checkout per the response to Question 2 in the RAI of May 2, 2006.

Thirteenth, 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 selection. In lines 4 through 6, the allowable fabrication methodology is changed to allow high purity uranium silicide-aluminum dispersion fuel but the currently allowed high purity aluminum-uranium alloy fuel option is removed. 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 of Appendix II of both submittals for the selected LEU fuel. In addition, on page 23 of the Tech Specs, in Section 5.3(1), the UFTR uranium-235 possession limit is raised to 9.00 kg to allow receipt of LEU fuel for conversion while HEU remains at the facility per the response to RAI Question 1 in Attachment II.

Also, 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 of Attachment II of the original submittal.

Fourteenth, on page 24 of the Tech Specs, in Section 5.5.1, Reactor Control System, in the second paragraph, the integral reactivity worths of the individual safety and regulating control blades are updated based upon the results of the LEU conversion analysis contained in Table 4-13 per the response to RAI Question 27 in Attachment II.

Fifteenth, on page 26, in Section 5.6.1, Primary Cooling System, the description of the instrumentation is returned to its status prior to Amendment 25 being approved so it delineates thermocouples at each fuel box outlet and the main inlet and outlet alarming and recording in the control room.

Finally, on page 38 of the Tech Specs, in Section 6.6.3, Other Special Reports, item (2) is augmented to include also reporting to the NRC significant changes in the transient or accident analysis for the HEU to LEU conversion just as with the previous analysis from the 1981 submittal per the response to RAI Question 28 in Attachment II.

These changes as submitted are considered to have minor safety significance. These proposed changes have been reviewed in progress by UFTR management and by the Reactor Safety Review Subcommittee (RSRS), as well as formally prior to submittal, with both concurring on this evaluation. In addition, as noted in the Attachment II safety analyses of the original submittal and the current Attachment II, a number of the members of the RSRS have been involved in developing this submittal.

This entire submittal consists of one signed original letter of transmittal plus Attachment I (Tech Spec changes) containing the proposed changes comprising the requested Amendment 26 (versus previous Amendment 25) to the UFTR Technical Specifications and Attachment II (safety analyses) containing details of supporting analyses per NUREG-1537 responding to the two Requests for Additional Information. Thirteen additional photocopied sets are enclosed.

We appreciate your consideration of this amendment. Please advise if further information is needed.

Sincerely,

William G. Vernetson

William G. Vernetson
Director of Nuclear Facilities

WGV/dms

Attachments I & II plus
Enclosures (13 sets of letter and attachments)

cc: Al Adams, NRC Project Manager
Craig Bassett, NRC Inspector
Reactor Safety Review Subcommittee

Sworn and subscribed this 20 day of June 2006.

Diana L. Dampier
Notary Public

Diana L. Dampier
Commission # DD452982
Expires July 20, 2009
Bonded Troy Fain - Insurance, Inc. 800-385-7019

ATTACHMENT I

TECH SPEC CHANGES

(includes all proposed changes for HEU to LEU conversion)

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.

(2) The specific resistivity of the primary coolant water shall not be less than 0.4 megohm-cm for periods of reactor operations over 4 hours.

Bases: Operating experiences and detailed calculations of Argonaut reactors and for the HEU to LEU conversion have demonstrated that Specification (1) suffices to maintain 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. Specification (2) suffices to maintain adequate water quality conditions to prevent deterioration of the fuel cladding and still allow for expected transient changes in the water resistivity.

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 set points.

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 kW.

(2) The primary coolant flow rate shall be greater than 36 gpm at all power levels greater than 1 watt.

(3) The average primary coolant
   (a) inlet temperature shall not exceed 109° F
   (b) 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 Channels 1 and 2 neutron chambers shall be 90% or more of the established normal value.

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

(7) The primary coolant flow rate shall be monitored at the return line.

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

(9) The secondary coolant flow shall satisfy the following 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 and the well water flow rate shall be larger than 60 gpm when using the well system for secondary cooling.

   or

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

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

(11) The reactor vent system shall be operating during reactor operations.

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

Bases: The University of Florida Training Reactor (UFTR) limiting safety system settings (LSSS) are established from operating experience and safety considerations. The LSSS 2.2.3 (1) through (10) 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 for the six fuel boxes, are monitored and recorded in the control room. LSSS 2.2.3 (11) are 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 from the cell. LSSS 2.2.3 (12) are established to protect reactor personnel from potential external radiation hazards caused by loss of biological shielding.

3.0 LIMITING CONDITIONS FOR OPERATIONS

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

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

(2) Excess Reactivity: The core excess reactivity at cold critical, without xenon poisoning, shall not exceed 1.4% Δk/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% Δk/k sec, when determined as an average over any 10 sec 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 3.5 of this report.

(6) Bases: These specifications are provided to limit the amount of excess reactivity to within limits known to be within the self-protection capabilities of the fuel, to ensure that a reactor shutdown can be established with the most reactive blade out of the core, to ensure a negative overall coefficient of reactivity, and to limit the reactivity insertion rate to levels commensurable with efficient and safe reactor operation.

3.2 Reactor Control and Safety Systems

3.2.1 Reactor Control System

(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) The reactor shall not be started unless the reactor control system is operable.

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

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

(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.

3.2.2 Reactor Safety System

(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.

3.2.3 Reactor Control and Safety Systems Measuring Channels

The minimum number and type of measuring channels operable and providing information to the control room operator required for reactor operation are given as follows:

Channel	No. Operable
Safety 1 and 2 power channel	2
Linear with auto controller	1
Log N and period channel*	1
Startup channel*	1
Rod position indicator	4
Coolant flow indicator	1
Coolant temperature indicator	
Primary	7
Secondary	1
Core level	1
Ventilation system	
Core vent annunciator	1
Exhaust fan annunciator	1
Exhaust fan rpm	1

* Subsystems of the wide range drawer

Table 3.1 Specifications for Reactor Safety System Trips

Specification	Type of Safety System Trip
Automatic Trips	
Period less than 3 sec	Full
Power at 119% of full power	Full
Loss of chamber high voltage (≥10%)	Full
Loss of electrical power to control console	Full
Primary cooling system	Rod-drop
Loss of pump power Low-water level in core (< 42.5") No outlet flow Low inlet water flow (< 36 gpm)	
Secondary cooling system (at power levels above 1 kW)	Rod-drop
Loss of flow (well water < 60 gpm, city water < 8 gpm) Loss of pump power	
High primary coolant average inlet temperature (≥109° F)	Rod-drop
High primary coolant average outlet temperature (≥155° F)	Rod-drop
Shield tank	Rod-drop
Low water level (6" below established normal level)	
Ventilation system	Rod-drop
Loss of power to dilution fan Loss of power to core vent system	
Manual Trips	
Manual scram bar	Rod-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, and after repair or deenergization caused by a power outage
10% reduction of safety channels high voltage	4/year (4-month maximum interval)
Loss of electrical power to console	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 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 power to vent system and dilution fan	4/year (4-month maximum interval)
Manual scram bar	With daily checkout

3.2.4 Bases

The reactor control system provides the operator with reactivity control devices to control the reactor within the specified range 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 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 facility or environment, problems in removal or handling after irradiation including containment, transfer, and eventual disposition. Guidance should be obtained from the ANS 15.1 Standard. Experimental apparatus, material, or equipment to be inserted in the reactor shall be reviewed to ensure noninterference 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, the 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 combined absolute reactivity worth of all movable or nonsecured experiments 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 less than that which could cause a positive 20-sec stable period.

(4) Explosive Materials

Explosive materials shall not be irradiated.

(1) The evacuation alarm is actuated automatically when two area radiation monitors alarm high (≥25 mrems/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.

3.7 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 the fuel storage conditions, and to establish fuel performance and fuel-handling specifications with regard to radiological safety considerations.

Specifications:

(1) The maximum 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 the fuel-handling procedures.

(3) Fuel elements exhibiting release of fission products because of cladding rupture shall, upon positive identification, be removed from the core. 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 removed and handled 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 fuel loading is based on the present fuel configuration. The reactor systems doe 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.

3.8 Primary Water Quality

Applicability: These specifications apply to primary cooling system water in contact with fuel 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 hr.

(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) 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 3.8.3(1), 3.8.3(2) and 3.8.3(5) are designed to protect the fuel element integrity and are based upon operating experience. At the specified quality, the activation products (of trace minerals) do not exceed acceptable limits. Specification 3.8.3(3) is designed to detect and identify fission products resulting from fuel failure and to fulfill reportability requirements pertaining to liquid wastes. Specification 3.8.3(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.

3.9 Radiological Environmental Monitoring Program

3.9.1 General

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 3.8 of these Technical Specifications.

4.2.6 Reactor Building Evacuation Alarm Surveillance

(1) The coincidence automatic actuation of 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 conditioning system and the reactor vent system shall be tested as part of the weekly checkout.

(3) Evacuation drills for facility personnel shall be conducted quarterly, at intervals not to exceed 4 months, to ensure that facility personnel are familiar with the emergency plan.

4.2.7 Surveillance Pertaining to Fuel

(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$1 kW) before the last two layers of concrete block shielding may be moved to reach the core area and 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 loading operations. The assignment of responsibilities and training of the fuel-handling crew shall be performed according to written procedures.

4.2.8 Primary and Secondary Water Quality Surveillance

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

(a) Primary water resistivity shall be measured during the weekly checkout by a portable Solu Bridge 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 Solu Bridge alarming in the control room, shall be larger than 0.5 megohm-cm at the outlet of the DM.

(2) 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.

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

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

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 of U-235 per fuel plate.

The UFTR facility license authorizes the receiving, possession, and use of

(1) up to 9.00 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 questions 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 elements shall conform to these 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	12.5 g U-235 nominal

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 top 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 sec and the insertion time under trip conditions is stipulated to be less than 1 sec. The reactor blade withdrawal interlock system prevents blade motion which will exceed the reactivity addition rate of 0.06% Δk/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 rod drives, transmit rod position information to the operator control console. Reactor shutdown also can 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% Δk/k depending on position in the core and individual characteristics. The regulating blade worth is about 0.6%–0.8% Δk/k. The rod worths, drive speeds, and drop-time values are sufficiently conservative to ensure compliance with the multirange pico-ammeter. The pico-ammeter sends a signal to one 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.

5.6 Cooling Systems

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 lb above normal in the system also will 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 outlet and the main inlet and outlet (eight total), alarming and recording in the control room

(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 secondary cooling system and a city water secondary cooling system. The well secondary cooling system is the main system used for removal of reactor

(e) abnormal and significant degradation in reactor fuel, or cladding, or both, coolant boundary, or containment 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

6.6.3 Other Special Reports

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

(1) permanent changes in facility organization involving Level 1, 2 or 3 personnel;

(2) significant changes in the transient or accident analyses as described in the UFTR Safety Analysis Report dated January 1981 and in the HEU to LEU fuel conversion analyses.

6.7 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, 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 shutdown and significant unplanned transients shall be maintained for a minimum period of 2 years.

6.7.1 Records To Be Retained for a Period of at Least Five 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

ATTACHMENT II

SAFETY ANALYSES

(Response to Requests for Additional Information)

REQUEST FOR ADDITIONAL INFORMATION
UNIVERSITY OF FLORIDA TRAINING REACTOR
DOCKET NO. 50-83

1. License conditions and Technical Specification 5.3. Please propose and justify license possession limits for the new low-enriched uranium (LEU) fuel if they are different than your current approved possession limits. Your answer may impact the possession limits given in Technical Specification (TS) 5.3. Will any additional changes be needed to the facility license to allow for conversion from high-enriched uranium (HEU) to LEU fuel?

Response
Technical Specification 5.3 currently allows 4.82 kg of contained Uranium-235. Current actual possession value for contained U-235 is [redacted] For the expected 26 full 14-plate bundles with 12.5 g of U-235 per fuel plate, the additional U-235 could be 4.55 kg plus a bit more, allowing for plate uncertainties. Therefore, assume 4.90 kg bringing the total new possession limit to 8.90 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
3.9	93%	Materials Test Reactor (MTR)-Type Fuel
5.0	< 20%	MTR-Type Fuel
0.2	Any %	Fission Chambers, Flux Foils and Other Forms Used in Connection with Operation of the Reactor

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

2. Section 1.5, Comparison with Similar Facilities Already Converted. It is stated that a surface treatment will be applied to the fuel. This is a design feature that was not evaluated in NUREG-1313, "Safety Evaluation Report related to the evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors." Will this surface treatment have any impact on the thermal-hydraulic analysis (e.g., impact fuel thermal-conductivity, hydraulic diameter, create hot spots if spalling occurs, etc.)? The reference cited in the conversion Safety Analysis Report (SAR) discusses the importance of maintaining primary coolant pH in a range of 5.4 to 6.0. Please propose and justify a TS to maintain pH within a proposed range or discuss why limits on primary coolant pH are not needed. The reference also discusses not letting primary coolant stagnate in the core and not draining the core if water or air quality is poor. Discuss any steps you take in these areas.

Response

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. Calculations performed to support the December 2005 submittal did not account for this effect, which at that time was about 0.01 C. Consequently changes in the peak clad and fuel temperatures are negligible, and the margins to ONB and to DNB are unaffected.

The measured pH levels for the UFTR primary coolant for the period July 2005 – May 2006 show pH levels consistently below 6.0 and frequently in the 4.3–5.0 range versus the 5.4–6.0 recommended in INL/EXT-05-00256 Revision 0 (reference 3 in the conversion submittal).

Per phone conversation with Dr. Gerard Hoffman at ANL (630-252-6683) on May 15, 2006, 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

As to the question of primary coolant (PC) stagnating in the core, such stagnation is not possible for the UFTR. The 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.

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. Currently, if water or air quality is poor, the water may be left running though these conditions would be considered unusual.

3. Section 4.5, Dynamic Design. Are any changes in fuel element location planned from the LEU reference core during the life of the core and if so, how does that affect the results of the safety analysis?

Response

We expect this need may occur toward the end of life, e.g., 20 years from now; so, it is not an issue for achieving criticality. 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 (see conversion SAR 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 Q3 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.

4. Section 4.5, Dynamic Design. The LEU core will age differently than the HEU core because of the addition of uranium-238 and the production of plutonium. Are the various calculated results given for the LEU core (e.g., kinetics parameters, reactivity coefficients, etc.) and the accident analysis results based on the most conservative values over the 87 MWD life of the fuel? If not, please provide these results or explain why changes in parameters over core life is not significant.

Response

Kinetics parameters and reactivity coefficients corresponding to an LEU core with fresh fuel and with a burnup of 87 MWD are shown in the table below. Further details are given in Appendix Q4.

Kinetics Parameters and Reactivity Coefficients for the UFTR LEU Core. Values in boldface type are used in accident analyses.

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%

Several of the accident analyses were redone using the above data for the fresh core and the depleted core. For the postulated transient with a rapid insertion of 0.6% Δk/k in the LEU core, assuming failure of the period trip, the peak temperature reached in the fuel and cladding was larger by about 0.5 °C using kinetics parameters and reactivity coefficients for the depleted core. A detailed comparison of results is provided in Table 13-3 of Appendix Q15.

5. Section 4.5.1, Calculation Model. In order to understand the engineering uncertainty factor assumed for the calculated power density (Table B-1) it is important to understand the validity of the MCNP5 model. Can the data in Table 4-9 and estimates of the uncertainties in measurements be used to determine an uncertainty factor with greater confidence than the one given in Table B-1?

Response
In our opinion, the data in Table 4-9 and estimates of uncertainties in the measurements cannot be used to reduce uncertainties incorporated in the hot channel factors.

6. Section 4.7.1 Fuel Assembly and Fuel Box Geometry. A form-loss coefficient K is used to represent local losses at the inlet, outlet and the grid plate. There are at least two types of coolant flow paths in the fuel box, gaps between the fuel plates, and the region between the fuel bundles (vertical bypass) plus the region between the fuel bundles and the fuel box (channel against fuel box). Explain the selection of the value(s) of K for the flow paths between the plates and outside the plates and justify that conservative value(s) were used in the analyses. Since the coolant channels between the fuel plates are not sealed by a channel box there will be mixing of coolant between the two types of coolant flow paths. Explain the impact of ignoring the diversion of heated coolant to the unheated flow channel (e.g. the vertical bypass region between fuel bundles).

Response
There are indeed multiple parallel flow paths through each fuel box that must be considered in the hydraulic analysis. For each path the flow passes first through the grid plate and then through the fuel assembly region. As explained in Appendix Q6, 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.

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.

7. 4.7.2 PLTEMP/ANL v2.14 Code Description. The reference (Reference 11) cited for the code PLTEMP/ANL v2.14 is for an earlier version that does not model laminar forced convection. Provide an updated reference that documents the implementation of the new models and the validation of the new models for application to plate type geometries.

Response

The PLTEMP/ANL code and manual have been updated. The reference is now:
Arne P. Olson and Kalimullah, A USERS GUIDE TO THE PLTEMP/ANL V3.0 CODE, Reduced Enrichment for Research and Test Reactor (RERTR) Program, Argonne National Laboratory, Internal Memorandum, May 16, 2006.

One-sided heat transfer can be modeled for first and last coolant channels in a fuel subassembly (this is important for laminar flow only)[1]:

Nu=4.86

The laminar flow heat transfer coefficient is computed and compared with the turbulent flow value. The largest heat transfer coefficient is then used. The ORNL laminar forced convection correlation is [2]:

Nu=7.63

Channel friction factors can now be computed for laminar flow, and for the transition between laminar and turbulent flow. The laminar friction factor is calculated from:

$f=96/Re$, for $0 < Re < 2200$

In the transition region between laminar and turbulent flows, the friction factor is computed by reciprocal interpolation as

$$f_{L,T} = (3.75 - 8250/Re)(f_{t,3000} - f_{l,2200}) + f_{l,2200} \qquad 2200 < Re < 3000$$

where $f_{l,2200}$ is the laminar factor at a Reynolds number of 2200, $f_{t,3000}$ is the turbulent friction factor at a Reynolds number of 3000.

Laminar flow friction factors were validated by hand checks on the computed friction values using code output for Reynolds number (and by hand verification that the Reynolds number itself was correct). Simple pipe flow problems that can be solved by other means have also been used as part of the hydraulic solution verification of PLTEMP. Heat transfer coefficients are edited by PLTEMP, making their verification a simple task.

[1] ORNL Monthly Progress Report, ORNL/ANS/INT-5/V19, Oak Ridge National Laboratory, October, 1989.
[2] W. M. Kays, Convective Heat and Mass Transfer, McGraw-Hill Book Co., New York, 1966. Table 8-2, p. 117.

8. 4.7.2 PLTEMP/ANL v2.14 Code Description. The placement of Table 4-20 appears to be out of place. There is no mention of the table in the text. Also it is not clear if the table is based on Table B-1 in Appendix B because values of hot channel factors from the two tables do not match. This table misplacement is probably also the reason that tables after Table 22 are misnumbered.

Response
The hot channel factors have been revised and were used in the final thermal-hydraulics calculations. See Tables 4-20 and 4-21 of Appendix Q8, which is a revision of Section 4.7.

9. 4.7.2 PLTEMP/ANL v2.14 Code Description. The thermal-hydraulic analyses did not include systematic uncertainties, such as measurements of reactor power and coolant flow rate, in the calculations. According to the SAR, "These systematic uncertainties will be included in the interpretation of the results." This does not seem to be the case so how was this done?

Response
Due to an oversight, the thermal-hydraulics calculations in the December 2005 conversion proposal did not include the effects of systematic uncertainties, such as measurements of reactor power and coolant flow rate. The treatment of these uncertainties is done through the selection of the limiting safety system settings as discussed in Section 4.7.3.3 of Appendix Q8.

10. 4.7.2 PLTEMP/ANL v2.14 Code Description. The thermal-hydraulic analysis assumed that power generation is uniform along the width of a fuel plate. Explain how the analysis accounts for the peaking of power density along the width of a fuel plate.

Response
The effect of the variation in power along the width of the fuel plate was investigated in detail. First, the MCNP code was used to determine the variation in power density along the width of the limiting fuel plate. The fuel plate was divided into 11 vertical strips of equal width. The relative power generated in each strip was determined. Second, a representation thermal test problem was solved in the following two ways: 1) with a one-dimensional hand-calculation that imitates the PLTEMP solution for a single fuel plate between two identical coolant channels and 2) with a computational fluid dynamics, CFD, solution. Refer to Appendix Q10. The CFD solution was implemented with the STAR-CD code and employed about 200,000 computational cells. Peak clad temperature is the figure of merit here because it determines the margin to the Onset of Nucleate Boiling (ONB). The two solutions, for an assumed 30° C coolant inlet temperature, produced essential the same value of peak clad temperature, i.e., 83.9° C for the one-dimensional solution and 83.3° C for the CFD solution.

The one-dimensional solution conservatively ignores the redistribution of heat along the width and the length of the fuel plate, particularly into the clad that frames the thin edges of the fuel meat. This effect is particularly important because the highest clad temperature is both near the top of the fuel meat and near a thin vertical edge of the fuel meat. Another mitigating factor is that coolant viscosity decreases with temperature. This causes higher coolant velocities and, therefore greater flow, in the vertical strips where the power is greater. Thus, the analysis does not need to explicitly account for the peaking of power density along the width of a fuel plate. The one-dimensional PLTEMP analysis is shown to be completely adequate even when the peak-to-average width-wise variation in plate power is greater than 1.3.

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.

11. 4.7.2 PLTEMP/ANL v2.14 Code Description. PLTEMP/ANL v2.14 requires the "specification of where the hot channel factors are to be applied. Explain the process and the calculations that were used to determine the limiting safety case with hot channel factors included.

Response

The process and the calculations that were used to determine the limiting safety case with hot channel factors are included in Appendix Q11. The table of hot channel factors is included below for convenience of review.

Hot Channel Factors

uncertainty	type of tolerance	effect on bulk ΔT, fraction	value	toler-ance	toler-ance, fraction	hot channel factors: 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.

12. Appendix B. One of the random uncertainties not explicitly represented in Appendix B, Table B-1 but is noted in Reference 25 is the fuel density from end-to-end along the axial length of the plate. Explain the significance/insignificance of this uncertainty for the LEU plate to be used in UFTR.

<u>Response</u>
All the required information are given in Appendix Q8. The engineering uncertainty factor related to the fuel density from end-to-end along the axial length is included into the fuel homogeneity (local) factor.

13. Section 4.7.3 Thermal-Hydraulic Analysis Results. The thermal-hydraulic parameters shown in Table 4-1 in the SAR do not agree with a similar tabulation in Table 4-22. Explain the differences between the two tables and why the maximum fuel temperature is higher for the LEU core in one table and lower in the other.

<u>Response</u>
The corrected Table 4-1 is included in Appendix Q13. The corrected Table 4-22 is included as Table 4-23 in Appendix Q8.

14. Section 12.4. Emergency Plan. For any emergency plan changes you want to make under the reactor conversion please submit the specific changes and their justification along with replacement Emergency Plan pages.

The necessary changes have been made and submitted in a separate transmittal directly to the NRC.

15. Section 13.0 Accident Analysis. The explanation of Table 13-1 says that coefficients with the smallest magnitude were used in the analysis in order to be conservative. Explain why the fuel temperature coefficient used is not the smallest value for the LEU core according to Table 4-18.

<u>Response</u>
The Doppler coefficient was updated. All of the analyses have been redone and are included in Appendix Q15.

16. Section 13.1.3 Sudden Insertion of the Maximum Allowed Excess Reactivity. In addition to peak clad temperature. The safety of the LEU core in a sudden insertion of the maximum allowed excess reactivity accident hinges on the total energy released in the transient that is scaled from the SPERT-I tests. What is the calculated value for the accident with a step insertion of reactivity?

<u>Response</u>
A value of 8.8 MWs was calculated for the UFTR LEU core, as shown in Table 13-5. The value is shown as < 8.8 MWs because the LEU core will have a much larger negative Doppler coefficient than the HEU SPERT-1 core. As a result, energy release in the LEU core will be smaller than estimated value using measured data for the HEU core. This explanation is included in the second paragraph on page 50 of the December 2005 submittal. Further, as concluded in the response to the next question, this analysis is not necessary.

17. Section 13.1.3 Sudden Insertion of the Maximum Allowed Excess Reactivity. In the analysis of energy deposition the reactivity feedback coefficient is based on an empirical correlation from the SPERT-I test results. How does this value differ from what would be calculated based on the UFTR core design and using MCNP5?

<u>Response</u>
The term "reactivity feedback coefficient" to describe the parameter "b" was used in the text immediately following equation (13.1) on page 49. Since the units of the parameter "b" are s^{-1}/MWs, the parameter "b" is better described as an "energy conversion coefficient".

For the HEU core, the maximum reactivity worth of all movable and unsecured experiments was set to the maximum allowed excess reactivity, i.e., 2.3%. UFTR has decided to limit the LEU core maximum reactivity worth of all movable and unsecured experiments to 0.6% dk/k. Because of this decision, we do not need to address the sudden insertion of the maximum allowable excess reactivity. Instead, the accident scenario studied in Section 13.1.1 is sufficient. Revised analyses are provided in Appendix Q15.

18. Section 13.3, Fuel Handling Accident. It is stated that in all cases, the reactor would be shutdown from power operation for at least three days before fuel handling operations would occur. Propose a technical specification that captures this condition or discuss why this limit is not needed.

<u>Response</u>
For both the Fuel Handling Accident (Q. 18.) and for the Maximum Hypothetical Accident (MHA) (Q. 21.), in Section 13.3, Tech Spec 4.2.7 (1) 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.

19. Section 13.3, Fuel Handling Accident and Section 13.4. Maximum Hypothetical Accident. The locations chosen for evaluations of doses from these accidents is not consistent with the guidance in NUREG-1537, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors, Format and Content," Section 13.2. The NRC staff does not support the use of site and urban boundaries as discussed in ANSI/ANS-15.7 (note that this is not an active standard). Please provide dose results and bases for the following locations: (1) occupational doses for staff inside the reactor cell from the initiation of the event until they evacuate the reactor cell, (2) non-occupational dose to the person with maximum exposure (this could be someone at the outside of the operations boundary but may be at a different location depending on the release point) and (3) non-occupational dose at the nearest residence (if it is closer than East Hall Housing).

Response

Revised data for the Maximum Hypothetical Accident (MHA) is shown below. Revised analyses for the Fuel Handling Accident and the MHA are provided in Appendix Q19. These analyses provide detailed dose results and the basis for the three locations stated in the question. A conservative five minute interval is considered for determination of occupational exposure. The location of the most exposed member of the public is at the west fenced area, 16.5 m from access door of the reactor building. East Hall is the nearest permanent residence, located 190 m from the outside wall of the reactor building.

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	8.3×10^{-5}	1.5×10^{-4}	4.0×10^{-4}
190	24	0.0157	0.0193	0.0218	1.3×10^{-5}	1.5×10^{-5}	1.6×10^{-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) Leak Rate (%Vol / hr)			Whole Body Dose (rem) Leak Rate (%Vol / hr)		
		10	20	100	10	20	100
16.5	2	0.109	0.199	0.522	7.8×10^{-5}	1.4×10^{-4}	3.7×10^{-4}
190	24	0.0193	0.0182	0.0205	1.3×10^{-5}	1.4×10^{-5}	1.5×10^{-5}

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

20. Section 13.3, Fuel Handling Accident. An assumption of a reactor cell leak rate of 0.2% volume/hour is used in this analysis. What is the basis for this leakage assumption? This type of assumption normally becomes a technical specification limiting condition of operation with a surveillance requirement to periodically (annually has been accepted by NRC) confirm the leak rate. Please discuss.

Response
See tables in the response to question #19 and Appendix Q19.

21. Section 13.4, Maximum Hypothetical Accident. It is not clear if there is an assumed 3-day period of reactor shutdown before shielding block handling would occur (similar to the fuel handling restriction in the fuel handling accident analysis). If so, please propose a technical specification that captures this condition or discuss why this limit is not needed.

Response
For both the Fuel Handling Accident (Q. 18.) and for the Maximum Hypothetical Accident (Q. 21.), in Section 13.3, Tech Spec 4.2.7 (1) 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. For further detail, see Appendix Q19.

22. Section 13.4.1 Radionuclide Inventories. In Table 13-11, the third and fifth columns are meant to be 2.7% of the previous column. However, the results for the LEU core are not correct, except for the I-135m case, even though the results for the dose calculation are correct. It appears that the HEU 2.7% column was copied into the LEU 2.7% column and never changed to reflect the 2.7% of the LEU inventory.

Response
Editorial correction is made. See Appendix Q19.

23. Appendix A.1, Determination of Material Composition, last paragraph. There appears to be an equation missing from the text where an error message appears. Please correct.

Response
Editorial correction is made. See Appendix Q23.

24. TableA.3-1. There appear to be formatting errors in the first footnote on this table. Please correct.

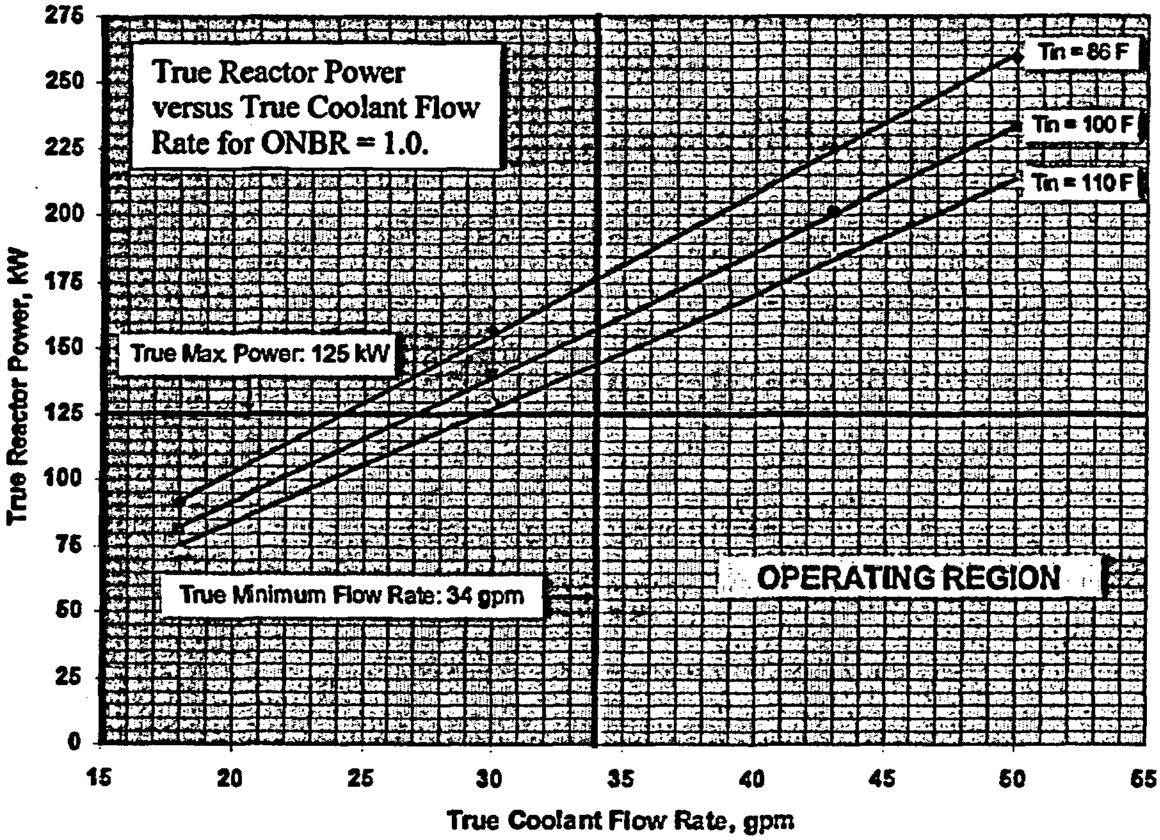
Response
Editorial correction is made. See Appendix Q24.

25. TS 2.1. Safety Limits. The regulations in 10 CFR 50.36 define safety limits (SLs) as 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. Exceeding a SL means that there is a possibility that the physical barrier has failed. That is the basis for requiring Commission approval for reactor restart if a SL is exceeded. As pointed out in the SAR, fuel cladding is the physical barrier of concern. It is stated that the integrity of the barrier is protected if the cladding temperature is kept below the incipient melting temperature of 582°C for the 6061-aluminum cladding of the LEU silicide fuel (however, NRC has accepted an upper temperature for aluminum-clad fuel of 530°C based on blister formation temperature-see NUREG-1313, "Safety Evaluation Report related to the evaluation of Low-Enriched Uranium Silicide-Aluminum Dispersion Fuel for Use in Non-Power Reactors," Section 3.3.3). You have based your SLs on the prevention of localized boiling. This results in SL process variable values significantly below those needed to protect against clad blister.

    As a result, in some cases, the accident analysis does not appear to support the technical basis for the SLs. For example, for the analysis of a step addition of reactivity of 0.6% dk/k into the core, the peak power for the LEU core with a SCRAM occurring is 320 kW. This would violate the SL in the proposed Technical Specification 2.1, namely, that "the power level shall not exceed 190 kW." Also, this result would indicate that the limiting safety system setting (LSSS) of 125 kW would not be in compliance with the regulations in 10 CFR 50.36(c)(1)(ii)(A) which states when a LSSS is specified for a variable on which a SL has been placed, the setting must be so chosen that automatic protective action will correct the abnormal situation before a SL is exceeded.

    Please propose revised SLs and LSSSs, as needed, that support the conclusions in your accident analyses for reactivity insertions and loss-of-coolant or explain why the proposed values in your SAR are acceptable.

Response
The value of 582°C will be changed to 530°C. This has no impact on the conclusions. The following figure provides information for selection of Limiting Safety System Settings (LSSS). Plotted in the figure are true reactor power versus true coolant flow rate for ONBR = 1.0 for three values of the coolant inlet temperature.



Based on the above figure, UFTR has decided on LSSS as follows: Power is 119 kW, coolant flow rate is 36 gpm, inlet temperature is 109 F, and outlet temperature is 155 F. These parameters were selected to account for uncertainties in measurements of the reactor power (±5%), coolant flow rate (±5%),, and coolant inlet/outlet temperature (±1°F). Detailed analyses are provided in Appendix Q8.

Further, the safety limit is based on maintaining integrity of the fuel; i.e., the fuel and clad temperature has to be less than 530 °C under any condition of operation. Appendix Q8 discusses how the LSSS protect the reactor fuel and clad from reaching this safety limit.

26. TS 5.3. In your conversion SAR only aspects of using U3Si2-Al dispersion fuel were discussed. In the proposed change to TS 5.3 it is stated: "The fuel matrix may be fabricated by alloying high purity aluminum-uranium alloy or the fuel matrix may be fabricated from uranium silicide-aluminum (U3Si2-Al) using the powder metallurgy process." Please explain why high purity aluminum-uranium alloy fuel remains in the technical specifications. If the high purity aluminum-uranium alloy were to be used, what effect would it have on all the safety analysis covered in the submittal? Is it possible that there could also be cores with a mix of the two types of fuel?

Response
The reference to aluminum-uranium alloy fuel was inadvertently not removed. There are no plans to have cores with a mix of the two fuels.

27. TS 5.5.1. This TS has integral worth values for the control rods. Does this information need to be updated based on calculated worths for control rods in the LEU core?

The calculated values from Table 4-13 for control blade integral reactivity worths for the LEU core are as follows:

Control Blade Type	LEU-fresh	LEU-depleted
Regulating	0.63%	0.66%
Safety 1	1.62%	1.65%
Safety 2	1.77%	1.76%
Safety 3	1.42%	1.46%

Therefore an updated range of about 0.6%–0.8% Δk/k is proposed for the Regulating Blade integral worth while 1.3%–2.0% Δk/k is the updated range for the Safety Blades.

28. TS 6.6.3(2). Should significant changes in the transient or accident analysis for the HEU to LEU conversion also be reported to NRC within 30 days of occurrence?

Response
It is agreed that significant changes in the transient or accident analysis for the HEU to LEU conversion should also be reported to the NRC.

29. Reactor Reload and Startup Plan. The conversion SAR states that existing procedures will be followed for reactor reload and startup. Provide a discussion of what measurements will be made and checked against calculated predictions (e.g. will initial criticality, excess reactivity, control rod worth, power calibration, etc. be measured and what will be the corresponding acceptance criteria?)

<u>Response</u>
A detailed discussion about the Reactor Reload and Startup Plan has been included in Appendix Q29.

Appendix Q3

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 Q3-1 compares the peak-to-average power ratio from the reference core (partial fuel bundle with 10 plates) with the two aforementioned cases.

Table Q3-1 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 Q3-2 gives the plate peak-to-average power ratio in the critical bundle 2-3.

Table Q3-2 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 Q3-2 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 Q3-1 shows the relative difference in the peak-to-average ratio power width profiles between the nomina; case (10 plates) and the two cases.

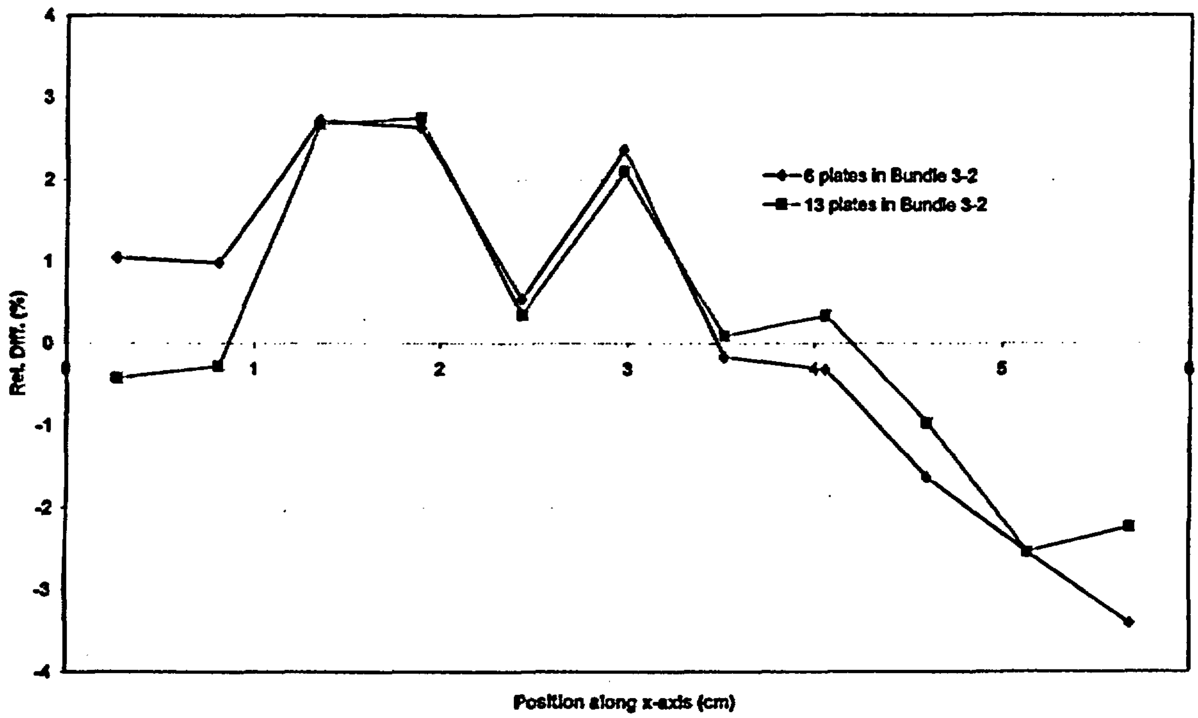


Figure Q3-1 Relative Difference between the Peak-to-Average Power Width Profile

The small changes in the power distribution presented in Table Q3-1,Table Q3-2 and Figure Q3-1 will have a minimal impact on the thermal-hydraulics analyses performed in the conversion submittal for the UFTR.

4.5.6.1 LEU Reactivity Coefficients

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/hydraulic-neutronics analyses for transients, the expected range of the coolant and fuel conditions should be taken into account when selecting the coefficients to employ.

Appendix Q6

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 Q6.1 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 Q6.1 is deemed to be a reasonable representation of the expected behavior of a symmetric quadrant of an LEU fuel box.

As Figure Q6.1 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:

$$\Delta P = (K + fL/D) \times \rho V^2/2 \qquad \text{(Q6.1)}$$

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 Q6.2 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 Q6.1 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:

$$K = \sum_i \left(K_i \left(\frac{A}{A_i} \right)^2 \right) \qquad \text{(Q6.2)}$$

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 Q6.2 and V in equation Q6.1 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 Q6.2 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 Q6.1 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 Q6.1, 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 Q6.1. 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 Q6.1 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.

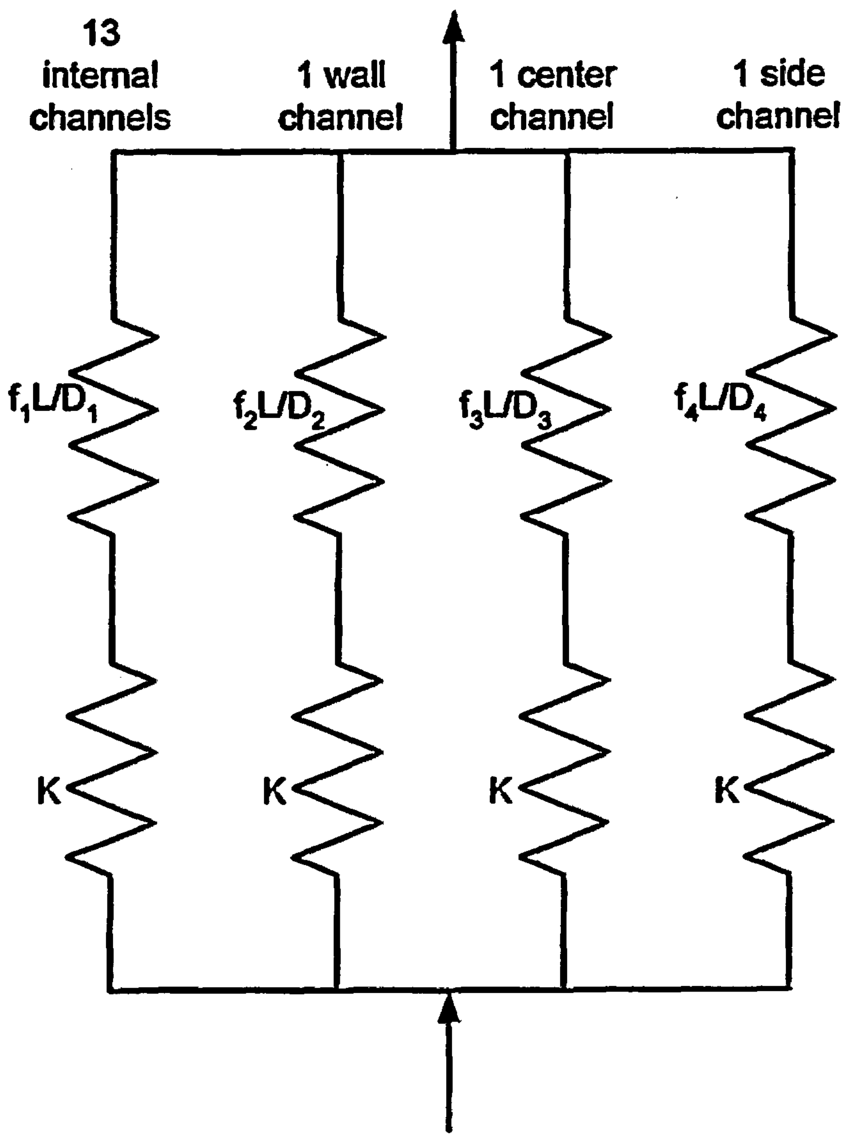


Figure Q6.1 Flow Network for One LEU Assembly Including Grid Plate

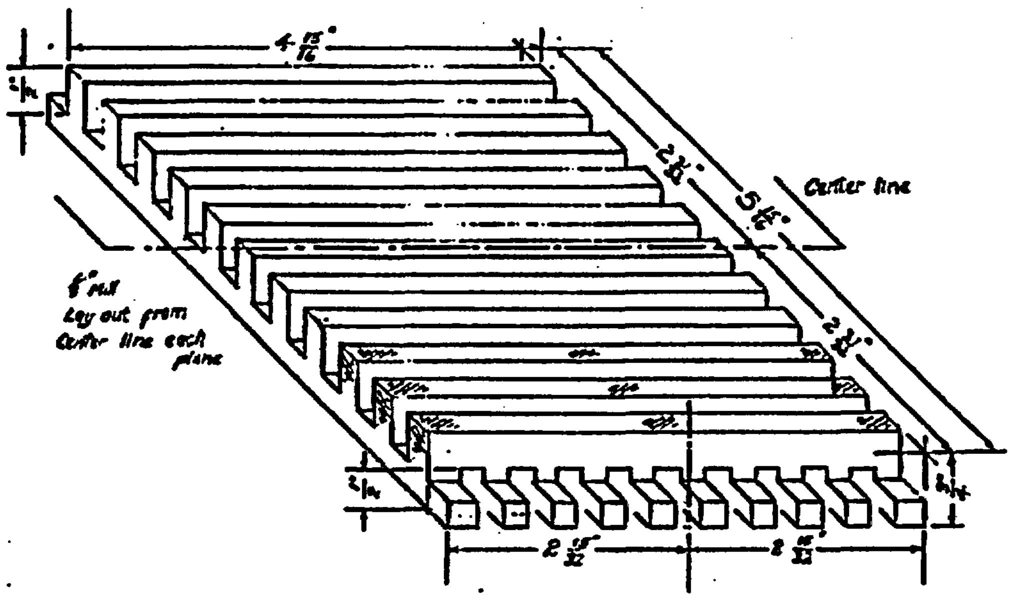


Figure Q6.2 Grid Plate

Table Q6.1 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

Appendix Q8

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

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. 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-hydraulic models and 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	5.125	130.2	5.125	130.2
Fuel box interior width	6.125	155.6	6.125	155.6
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. 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. Additional information is provided in Appendix Q6.

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 was obtained for a single LEU fuel plate with and without a power profile along the width of the fuel (see Appendix Q10). 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 Q11.

However, 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 4.7.3.3.

Table 4-20 Hot Channel Factors for the HEU Core

uncertainty	type of tolerance	effect on bulk ΔT, fraction	value	toler-ance	toler-ance, fraction	Hot Channel Factors			
						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

uncertainty	type of tolerance	effect on bulk ΔT, fraction	value	toler-ance	toler-ance, fraction	Hot Channel Factors			
						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 12.5 ± 0.35 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.

4.7.3 Thermal-Hydraulic Analysis Results
In this section, the PLTEMP/ANL V 3.0 code 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.

4.7.3.1 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

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.

4.7.3.2 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. 20) 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.

4.7.3.3 Limiting Safety System Settings for the LEU Core
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*.

Tin, F	86				100				110			
Tin, C	30				37.8				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.

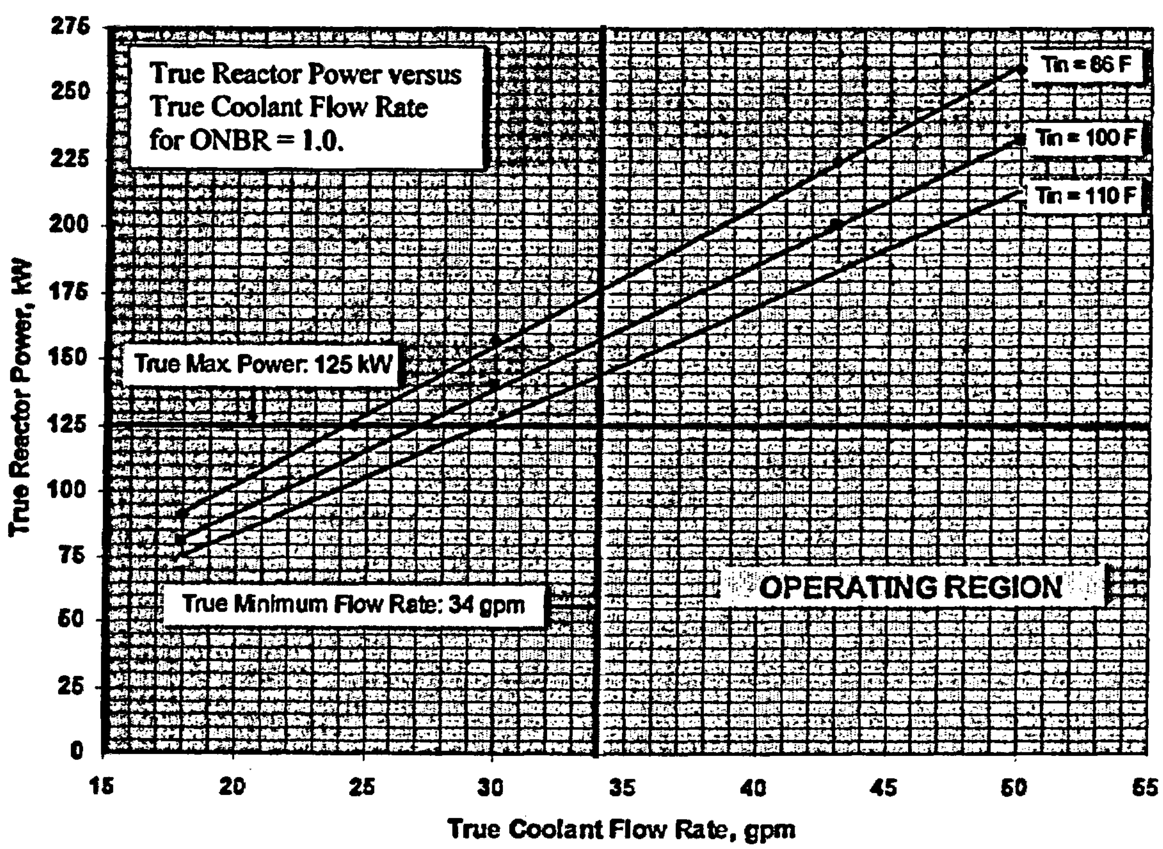
Figure 4-20 also shows horizontal and vertical solid lines that represent the true maximum power level of 125 kW and the true minimum coolant flow rate of 34 gpm. The true maximum value of the average coolant inlet temperature is 110 °F.

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



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.

References

11. Arne P. Olson and Kalimullah, "A User's Guide To The PLTEMP/ANL V3.0 Code", Reduced Enrichment for Research and Test Reactor (RERTR) Program, Argonne National Laboratory, May 16, 2006.

Appendix Q10

Figure Q10.1 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.[1] 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.

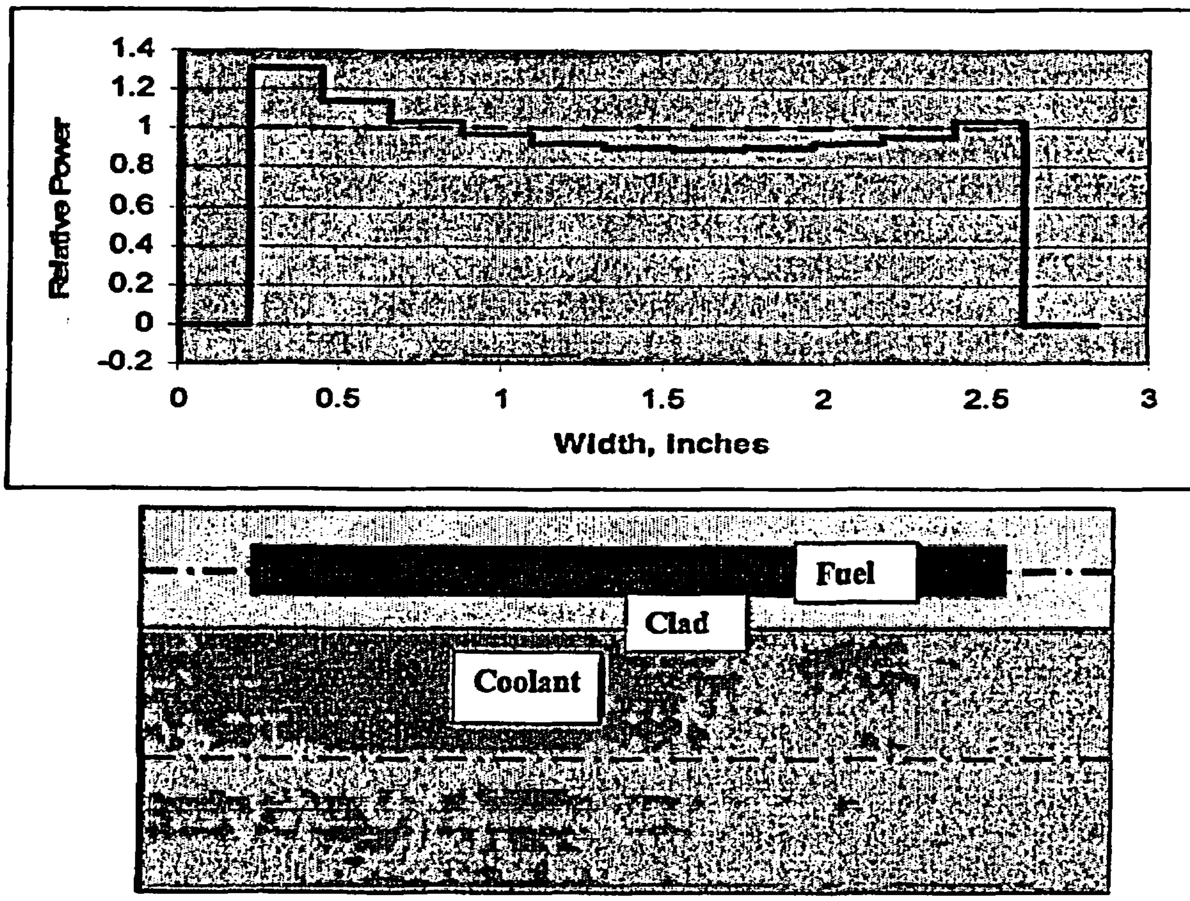


Figure Q10.1 Top View of Fuel Plate and Coolant Channel and Power Distribution along Hottest LEU Plate

[1] Subsequently, the MCNP calculation was redone causing a slight change in the power distribution from that shown in Figure Q10.1. 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 Q10.2 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 Q10.1, 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 Q10.1. The thin edges of the channel and the fuel plate were assumed to be insulated.

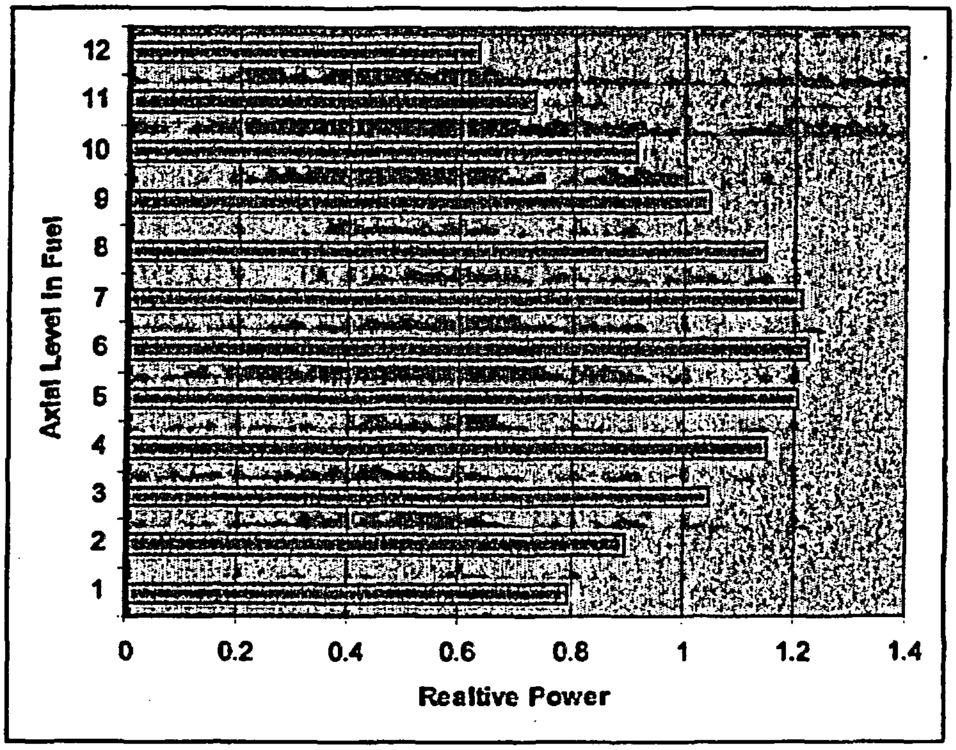


Figure Q10.2 Axial Power Distribution of Hottest LEU Fuel Plate

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 Q10.2 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 Q10.1 and extending the entire length of the fuel plate was divided into about 200,000 computational cells. Figure Q10.3 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 Q10.4 shows the CFD clad outer surface temperature distribution for this uniform power case.

As Figure Q10.4 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 Q10.1 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 Q10.1 Summary of Results

Power distribution	Peak Clad Temperature (°C)	
	PLTEMP	CFD Solution
2-D	83.9	83.3
Uniform	86.2	84.3

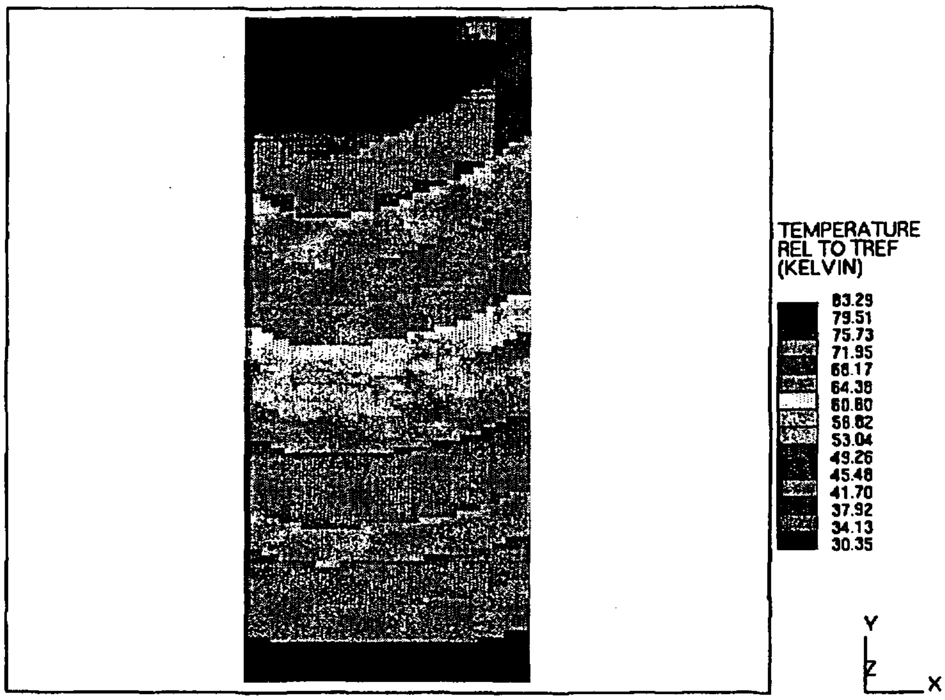


Figure Q10.3 Clad Surface Temperature Distribution Obtained from CFD Solution

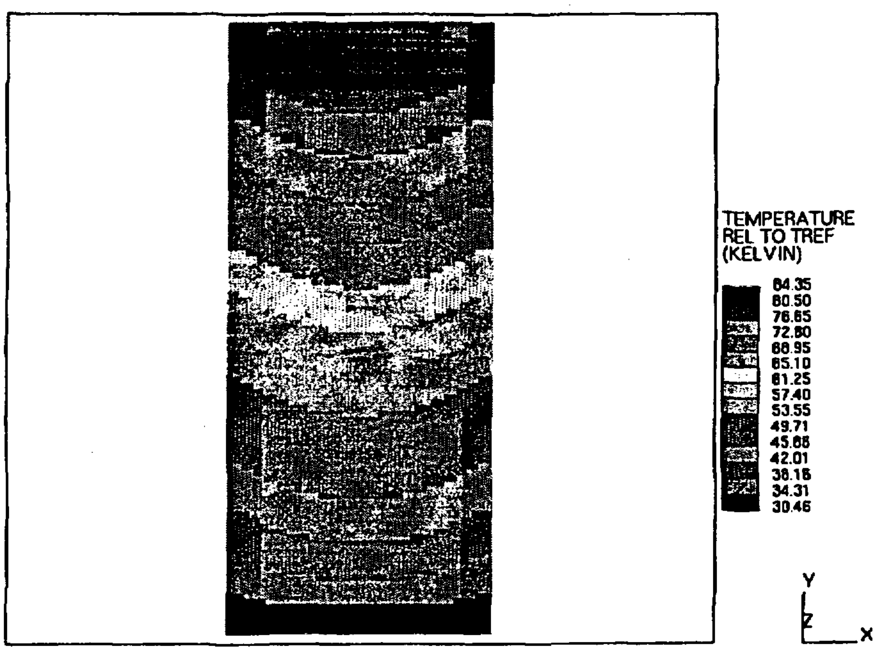


Figure Q10.4 Clad Surface Temperature Distribution Obtained from CFD Solution for Uniform Power Distribution

Appendix Q11

I. The PLTEMP Code

PLTEMP is designed to do steady-state thermal-hydraulic analysis of plate type research reactor cores. A single fuel assembly, multiple assemblies, or an entire core may be represented. Although all of the assemblies can be hydraulically coupled, heat transfer from one assembly to its neighbors is not represented in the model. The core is divided into a series of axial levels. For each axial level the code determines both the bulk coolant temperature in each coolant channel and the clad surface temperatures and heat fluxes on each side of each fuel plate. All of the individual heat transfer relationships used in the code are spatially one-dimensional. Temperature variations along the width of the fuel plates are not considered. At each axial level the code determines the peak fuel meat temperature and the location of the peak temperature within the fuel meat thickness.

In addition to determining all of the needed coolant, clad, and fuel meat temperatures and fuel plate heat fluxes, the code also evaluates the limiting criteria for onset of nucleate boiling, flow instability, and critical heat flux and compares the calculated plate temperatures and heat fluxes to them.

II. 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

$$\Delta T_b = \frac{q''}{h} \quad (3)$$

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}.

III. 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.

IV. 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 Q11.1. Two additional hot channel factors, not included in References 1 and 2, 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 Q11.1 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 Q11.1 represent the total combination of hot channel factors, they are not used in the PLTEMP code.

A line-by-line description of Table Q11.1 follows:

Fuel meat thickness (local)
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 Q11.1 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 Q11.1 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 2. 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 Q11.1. 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 1 and 2, 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 Q11.2 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 Q11.1 is informative.

V. 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, as requested by Question 11, the process and the calculations that were used to determine the limiting safety case with hot channel factors included have been explained.

1. R. S. Smith and W. L. Woodruff, *A Computer Code, Natcon, for the Analysis of Steady-State Thermal-Hydraulics and Safety Margins in Plate-Type Research Reactors Cooled by Natural Convection*, ANL/RERTR/TM-12, Argonne National Laboratory, Argonne Illinois, December 1988.

2. W. L. Woodruff, *Evaluation and Selection of Hot Channel (Peaking) Factors for Research Reactor Applications*, ANL/RERTR/TM-28, RERTR Program, Argonne National Laboratory, Argonne, Illinois, February 1997 [http://www.rertr.anl.gov/METHODS/TM28.pdf.].

Table Q11.1 Hot Channel Factors

uncertainty	type of tolerance	effect on bulk ΔT, fraction	value	toler-ance	toler-ance, fraction	hot channel factors			
						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.

Appendix Q13

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

	HEU	LEU
DESIGN DATA		
Fuel Type	U-Al alloy	U_3Si_2-Al
Fuel Meat Size		
Width (cm)	5.96	5.96
Thickness (cm)	0.102	0.051
Height (cm)	60.0	60.0
Fuel Plate Size		
Width (cm)	7.23	7.23
Thickness (cm)	0.178	0.127
Height (cm)	65.1	65.1
Cladding	1100 Al	6061 Al
Cladding Thickness (cm)	0.038	0.038
Fuel Enrichment (nominal)	93.0 %	19.75%
"Meat" Composition (wt% U)	[redacted]	62.98
Mass of ^{235}U per Plate (nominal)	[redacted]	12.5 g
Number of Plates per Fuel Bundle	11	14
Number of Full Fuel Bundles (current/expected)	21	22
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

THERMAL-HYDRAULIC PARAMETERS (100kW, 43 gpm, 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

[2] At nominal operating conditions

Appendix Q15

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% $\Delta k/k$. [Section 3.1(2)]
- The absolute reactivity worth of any single moveable or nonsecured experiment shall not exceed 0.6% $\Delta k/k$. [Section 3.5(3a)]
- The total absolute reactivity worth of all experiments shall not exceed 2.3% $\Delta k/k$. [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% $\Delta k/k$. [Section 3.1(2)]
- The absolute reactivity worth of all moveable or nonsecured experiment shall not exceed 0.6% $\Delta k/k$. [Section 3.5(3a)]
- The total absolute reactivity worth of all experiments shall not exceed 1.4% $\Delta k/k$. [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% $\Delta k/k$ 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% $\Delta k/k$/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 (Sections 4.5.3.1 and 4.5.6.1) 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} ($\Delta\rho$/°C)	-5.91E-05	-5.68E-05	-5.26E-05
C_{fuel} ($\Delta\rho$/°C)	-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.[1] 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	1.016	0.508
Fuel plate thickness, mm	1.778	1.270
Fuel plate width, mm	72.26	72.26
Fuel plate heated length, mm	600.0	600.0
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% $\Delta k/k$ 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.

References

1. The RELAP5-3D Code Development Team, "RELAP5-3D Code Manual", INEEL-EXT-98-00834, Revision 2.2, Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho (October 2003).

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.

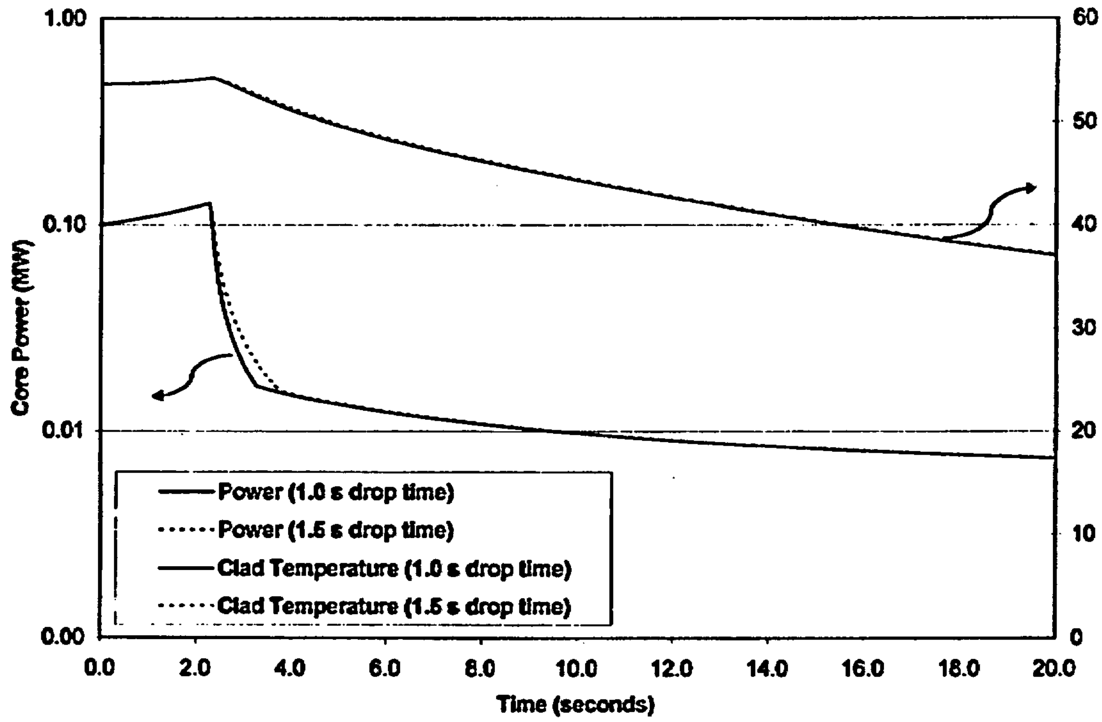


Figure 13-1 Slow Insertion of 0.06% Δk/k/s with SCRAM in HEU Core

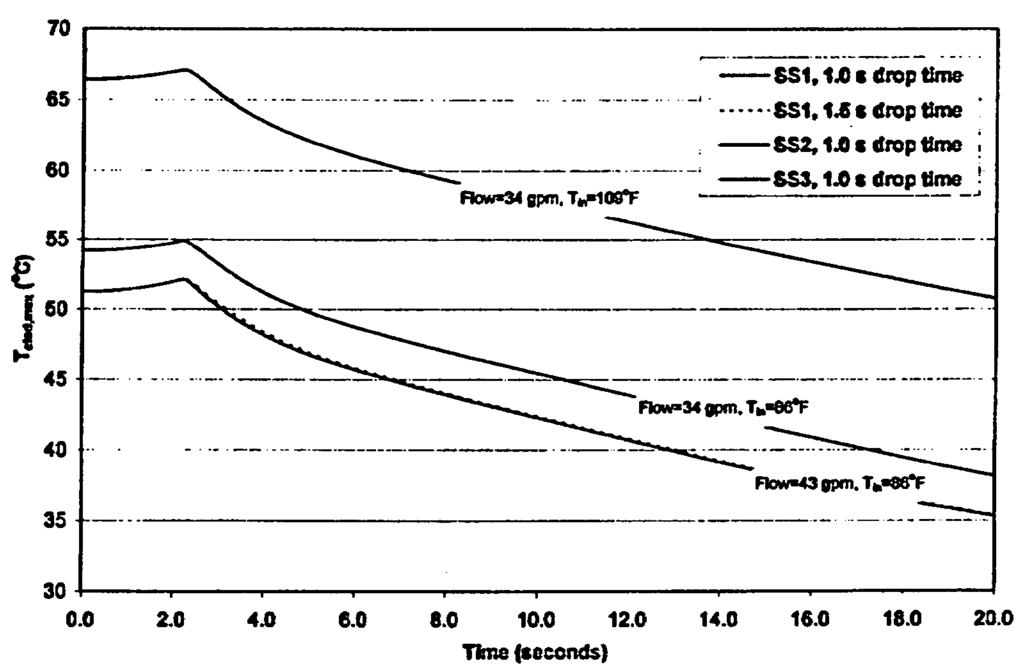


Figure 13-2 Slow Insertion of 0.06% Δk/k/s with SCRAM in Depleted LEU Core

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.

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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.

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. 17) 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 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				
Kr 85m				
Kr 88				
I 130				
I 131				
I 132				
I 133				
I 135				
Xe 133				
Xe133m				
Xe 135				

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. 20), 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. 21) 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.17). 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. 23) 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. 21) 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 Reference 26. 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. 22
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. 22
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. 23) based on inhalation.
8. Dose coefficients for Xenon and Krypton taken from FGR No. 12 (Ref. 24) 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. 22) 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 Reference 26. 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. 22
2. Breathing rate: 1.75×10^{-4} m^3/sec for 8 to 24 hours based on Ref. 22
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. 22
6. Reactor cell leak rate: 10% to 20% volume / hour.
7. Dose coefficients for Iodine taken from Federal Guidance Report (FGR) No. 11 (Ref. 23) based on inhalation.
8. Dose coefficients for Xenon and Krypton taken from FGR No. 12 (Ref. 24) 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. 22). 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. 27). 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. 22 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. 28, 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. 28 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.17).

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 References 23 and 24. The half life values are given for each isotope and are used to determine the effect of the reactor cell leak rate. Reference 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-m^3)
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	Time of Exposure	Thyroid Dose (rem)			Whole Body Dose (rem)		
		Leak Rate (%Vol / hr)			Leak Rate (%Vol / hr)		
(m)	(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	Time of Exposure	Thyroid Dose (rem)			Whole Body Dose (rem)		
		Leak Rate (%Vol / hr)			Leak Rate (%Vol / hr)		
(m)	(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				
Kr 85m				
Kr 88				
I 130				
I 131				
I 132				
I 133				
I 135				
Xe 133				
Xe 133m				
Xe 135				
Xe 135m				
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 Kr 85m Kr 88 I 130 I 131 I 132 I 133 I 135 Xe 133 Xe 133m Xe 135 Xe 135m				
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 Kr 85m Kr 88 I 130 I 131 I 132 I 133 I 135 Xe 133 Xe 133m Xe 135 Xe 135m				
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 [redacted] 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 [redacted] concrete shield block would result in severe mechanical damage to the fuel and a significant release of fission products.

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. 17) 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				
Kr 85m				
Kr 88				
I 129				
I 130				
I 131				
I 132				
I 133				
I 135				
Xe 133				
Xe 133m				
Xe 135				
Xe 135m				

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	8.3×10^{-5}	1.5×10^{-4}	4.0×10^{-4}
190	24	0.0157	0.0193	0.0218	1.3×10^{-5}	1.5×10^{-5}	1.6×10^{-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	7.8×10^{-5}	1.4×10^{-4}	3.7×10^{-4}
190	24	0.0193	0.0182	0.0205	1.3×10^{-5}	1.4×10^{-5}	1.5×10^{-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				
Kr 85m				
Kr 88				
I 130				
I 131				
I 132				
I 133				
I 135				
Xe 133				
Xe 133m				
Xe 135				
Xe 135m				
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				
Kr 85m				
Kr 88				
I 130				
I 131				
I 132				
I 133				
I 135				
Xe 133				
Xe 133m				
Xe 135				
Xe 135m				
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				
Kr 85m				
Kr 88				
I 130				
I 131				
I 132				
I 133				
I 135				
Xe 133				
Xe 133m				
Xe 135				
Xe 135m				
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.

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Appendix Q23

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,

$$mC_p(T - T_{init}) = P\,t \qquad \text{(A.1.1)}$$

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.

Appendix Q24

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)	Δk/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).

Appendix Q29

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 1 and Table 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 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 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 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 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 1

UFTR LEU CORE FUEL LOADING FROM 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 1 will be used to record actual loading.

UFTR LEU CORE FUEL LOADING FROM 6M DRUMS (Date: __________)

Table 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 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).