CORE CONVERSION ANALYSES
FOR THE
PORTUGUESE RESEARCH REACTOR

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ABSTRACT

Design and safety analyses are presented for conversion of the Portuguese Research Reactor (RPI) from the use of HEU fuel to the use of LEU fuel. The analyses were performed jointly by the RERTR Program at the Argonne National Laboratory (ANL) and the Instituto Tecnológico e Nuclear (ITN). The LEU fuel assembly design uses U_3Si_2-Al dispersion fuel with 4.8 g U/cm^3 and is very similar to the HEU fuel design. The results of neutronic studies, steady-state thermal-hydraulic analyses, accident analyses, and revisions to the Operating Limits and Conditions demonstrate that the RPI reactor can be operated safely with the new LEU fuel assemblies. Delivery of the LEU fuel is expected around the end of 2006, with conversion in early 2007. The HEU fuel is planned to be returned to the US in 2008.

INTRODUCTION

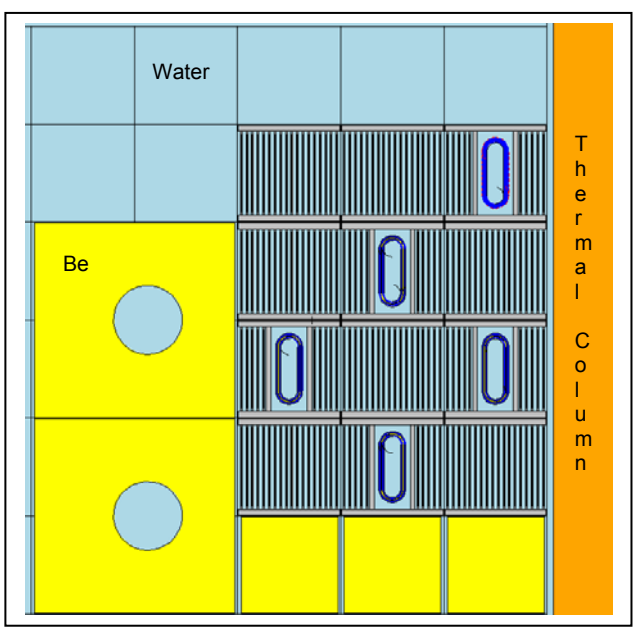
The purpose of this study was to complete the reactor design, performance and safety analyses that would enable conversion of the reactor fuel from HEU to LEU without requiring any changes to the control rods, control rod drive mechanisms, the instrumentation and control system, or other reactor components. A feasibility study[1] was completed in 2005. Documents that were reviewed by ANL as bases for the design and safety evaluations were the RPI Operating Limits and Conditions (OLC), design drawings, and historic analyses of the facility. All of the additional information and data needed to construct the reactor models and perform the analyses were provided by ITN. The methods and codes that were utilized have been qualified by extensive conversion analysis experience and international benchmark.

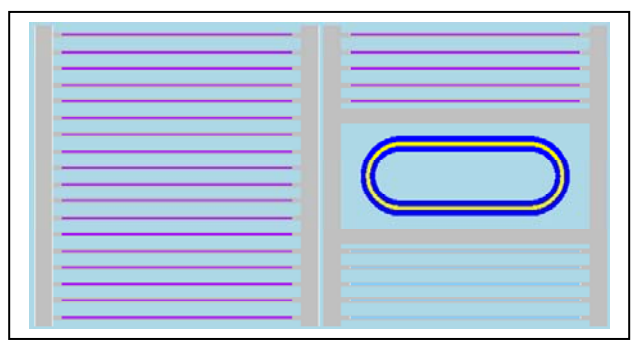
Only those reactor parameters and safety analyses which could change as a result of replacing the HEU fuel with LEU fuel are addressed. Table 1 provides a summary of the key design features of the HEU and LEU fuel assemblies and the initial HEU and LEU cores. LEU silicide fuel[2] with uranium densities up to 4.8 g U/cm^3 is currently used in 21 research reactors with power levels up to 70 MW, located in 13 countries including Australia, Canada, France,

Germany, Japan, and the U.S. Figure 1 shows the MCNP[3] model of RPI's initial HEU core configuration P1/2 selected as the reference core for these analyses.

Table 1. Summary of HEU and LEU Design Data

DESIGN DATA	HEU	LEU
Fuel Type	MTR Plate	MTR Plate
Fuel "Meat" Composition	U-Al alloy	U_3Si_2-Al
Uranium Enrichment (nominal)	93.2 %	19.75%
Fuel Plate Thickness (mm) Width (mm) Length (mm)	1.27 71.0 625.5	1.37 71.0 625.5
Fuel Meat Thickness (mm) Width (mm) Length (mm)	0.5 63.4 596.9	0.6 63.4 596.9
Uranium Density in Fuel Meat, g/cm^3	0.83	4.8
Cladding Material	1100 Al	AG3 NE
Cladding Thickness (mm)	0.38	0.38
Number of standard assemblies (Initial Core) Mass of ^{235}U per standard assembly Number of plates per standard assembly Coolant Channel Thickness (mm)	7 265 18 3.15	7 376 18 3.05
Number of control assemblies (Initial Core) Mass of ^{235}U per control assembly Number of plates per control assembly Coolant Channel Thickness (mm)	5 147 10 3.15	5 209 10 3.05





Standard Fuel Assembly Control Fuel Assembly

Above: Enlarged view of a standard fuel assembly with 18 fuel plates (left) adjacent to a control fuel assembly with 10 fuel plates and an oval-shaped shim-safety control rod (right) used in the MCNP model.

Figure 1. MCNP Model of RPI's Initial HEU Core Configuration P1/2 Selected as Reference.

The HEU design and the LEU design each contain 18 fuel plates per standard assembly and 10 plates per control assembly. The LEU fuel assembly design contains plates with a fuel meat thickness of 0.6 mm to replace the HEU fuel which has a fuel meat thickness of 0.5 mm. This results in a ^{235}U loading of 376 g per LEU standard assembly compared with 265 g per HEU standard assembly.

LEU conversion of the RPI core requires no changes to the control rods, control rod drive mechanisms, instrumentation and control system, or other reactor components except for the fuel.

Neutronics Analyses

Monte Carlo burnup calculations using the REBUS-MCNP code[3,4] were done for sequences of HEU and LEU core configurations shown in Figure 2. These studies concluded that the selected LEU fuel assembly design would allow operation of the reactor for 502 MWd over a period of ten years. The 500 MWd design goal was achieved utilizing a total of 14 LEU fuel assemblies in four core configurations. After 500 MWd, the average ^{235}U burnup of the LEU core was about 14% of the initial ^{235}U. The highest-burned assembly had an average ^{235}U burnup of about 22%.

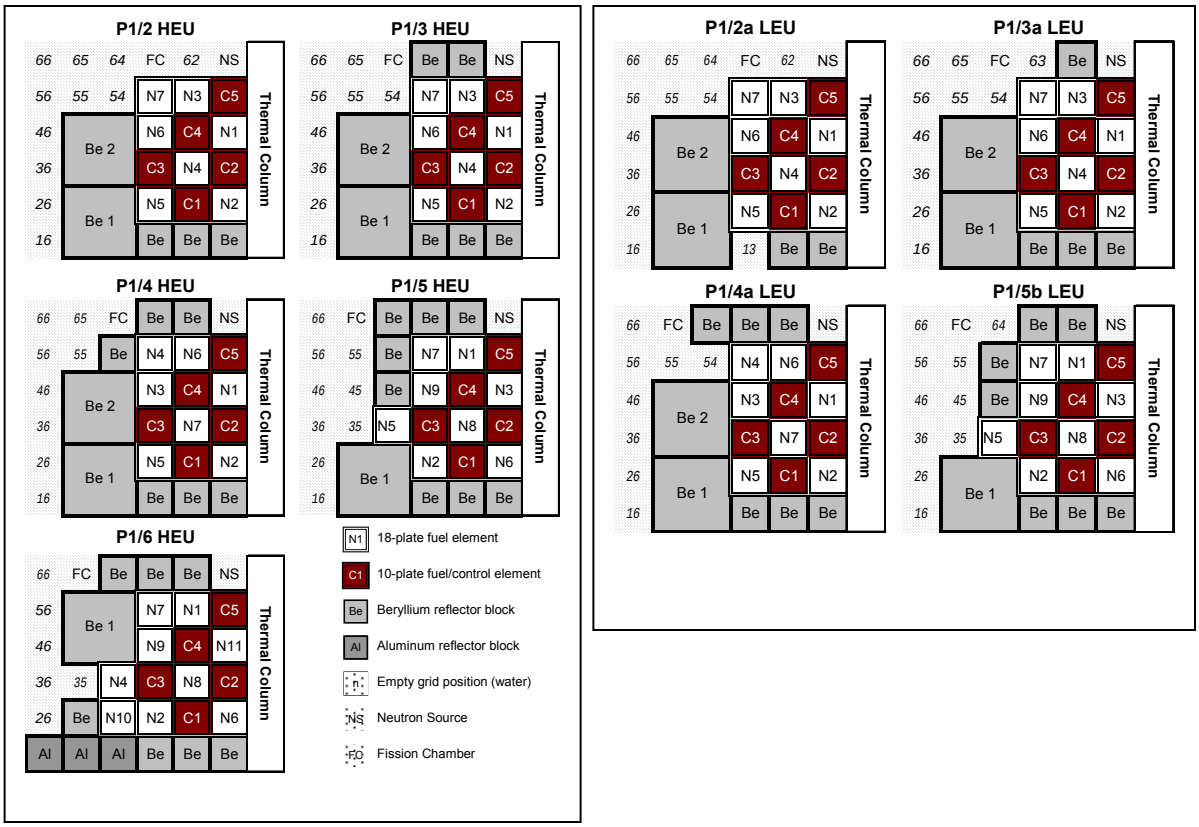


Figure 2. Core Configurations for the HEU and LEU Burnup Calculations.

The control rods were modelled in great detail. Calculations were performed for 53 measured critical and perturbed rod configurations of the HEU reference core in order to compare measured and calculated values of criticality, and the incremental and integral rod worths of two

key control rods and the regulating rod. The calculated and measured results are in good agreement for the HEU core. The RMS bias among the critical configurations was 0.4% Δk/k and the integral worths were within 0.5% Δ k/k. Shutdown margins for each of the HEU and LEU cores exceed requirements of the Operating Limits and Conditions by a wide margin. Values for the initial and ending HEU and LEU cores are shown in Table 2.

Table 2. Key Reactor Parameters at BOC for the Initial and Ending HEU and LEU Cores

REACTOR PARAMETERS	HEU		LEU	
Core	**P1/2 (Fresh)**	**P1/5 (Burned)**	**P1/2a (Fresh)**	**P1/5b (Burned)**
Excess Reactivity (% Δk/k)	4.40	5.16	3.18	4.09
Minimum Shutdown Margin (% Δk/k)	- 6.58	- 3.72	- 6.71	- 4.40
Effective Delayed Neutron Fraction	0.00795 ± 0.00014	0.00808 ± 0.00012	0.00800 ± 0.00012	0.00800 ± 0.00012
Prompt Neutron Lifetime (μs)	64 ± 4	66 ± 4	55 ± 4	61 ± 4
Average Coolant Void Coefficient (% Δk/k/%void)	-0.21 ± 0.014	-0.22 ± 0.014	-0.26 ± 0.015	-0.20 ± 0.014
Coolant Temperature Coefficient (% Δk/k/^{o}C)	-0.013 ± 0.0001	-0.014 ± 0.0001	-0.009 ± 0.0001	-0.009 ± 0.0001
Fuel Temperature Coefficient (% Δk/k/^{o}C)	0 ± 0.0001	0 ± 0.0001	-0.001 ± 0.0001	-0.001 ± 0.0001

Values of the kinetics parameters and reactivity coefficients shown in Table 2 are very similar for the HEU and LEU cores with fresh or depleted cores. Detailed power distributions for use in the steady-state thermal-hydraulics and accident analyses were computed using MNCP on a plate-by-plate basis with 20 axial nodes. Power peaking factors were similar for the HEU and LEU cores.

Thermal neutron fluxes in key experiment positions are compared in Table 3 for the HEU and LEU cores. Most of the large differences are due to differences in the HEU and LEU core configurations.

Table 3. Ratio of MCNP4C Thermal Fluxes: LEU/HEU at each BOC.

Core	Maximum Single Grid Position		Be Experiment Block Ratio of Predicted Flux		Beam Tube E4 Entry Ratio of Predicted Flux
	Position(s)	Ratio of Predicted Flux	Be 1 Grid Rows 1 & 2	Be 1 Grid Rows 3 & 4	
Thermal Flux, neutron energy < 0.625 eV, in maximum 5 cm segment of tube					
P1/2a LEU vs. P1/2 HEU	LEU 13, HEU 54	107%	84%	91%	83%
P1/3a LEU vs. P1/3 HEU	54	85%	99%	92%	102%
P1/4a LEU vs. P1/4 HEU	LEU 54, HEU 55	217%	93%	88%	102%
P1/5b LEU vs. P1/5 HEU	35	95%	95%	*na*[1]	100%

[1] *na* indicates a position filled by a fuel assembly or Be block, thus unavailable as an experimental location.

Thermal-Hydraulics Analyses

Detailed thermal-hydraulics analyses were done for steady-state operation under both forced-flow using the PLTEMP/ANL code[5] and natural convection cooling conditions using the NATCON code[6]. The first and principal barrier protecting against the release of radioactivity is the cladding of the fuel plates. The safety limit specified in the OLC requires that fuel and cladding temperatures do not exceed 530 ^{o}C, which is about 20 ^{o}C below the measured temperature[2] for initiation of fission product release. For normal steady-state operation, the only mechanism for approaching failure of the cladding is a flow instability that can result from low-pressure boiling.

As shown in Figure 3, for normal operation with forced-convection flow, the power level at which flow instability could occur was calculated to be 3.3 MW and could be as low as 2.8 MW for the selected safety system settings (SSS) in the OLC if the actual coolant flow rate were to reach 3.0 m^3/min. In the natural convection mode, the maximum nominal power is 67 kW and the SSS is 100 kW. Analyses show that flow instability could not occur until a power level of about 250 kW is reached. The thermal hydraulics analyses demonstrate that the RPI can be operated with adequate margins of safety in both the steady-state forced convection and natural convection modes of operation.

The safety system settings that were selected 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.

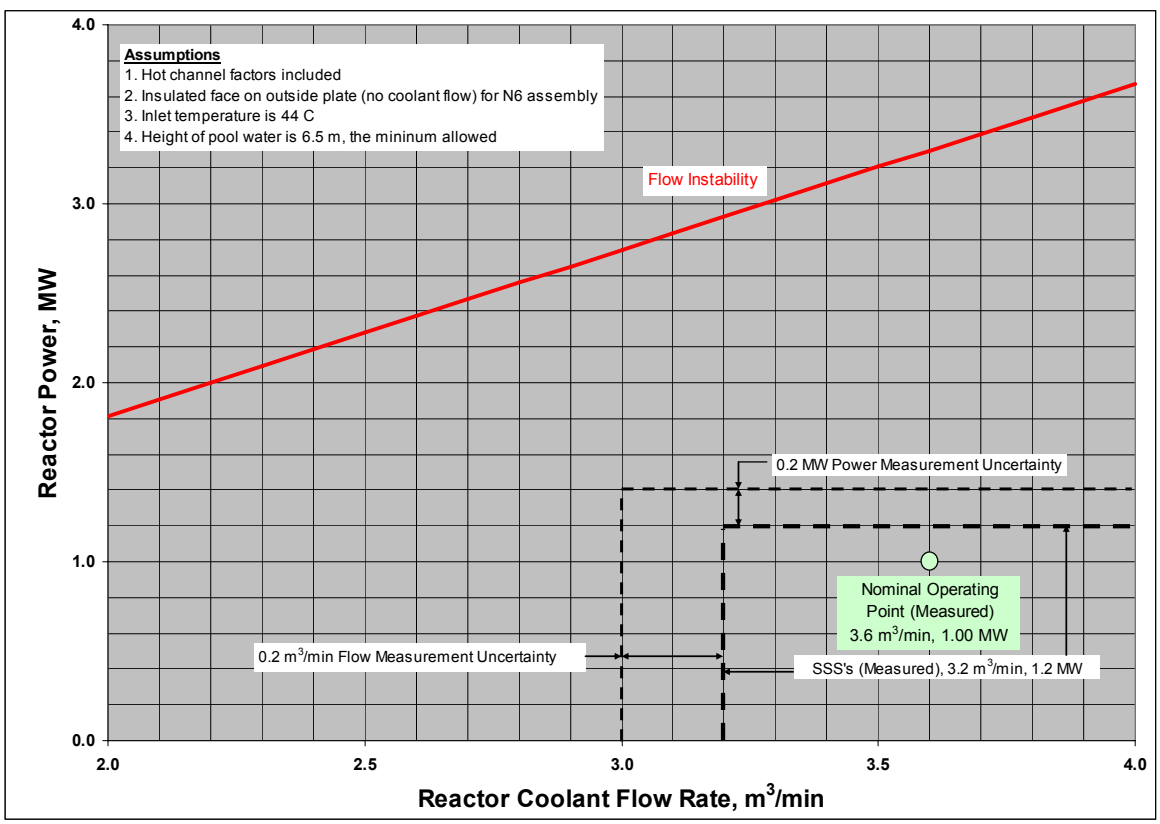


Figure 3. Reactor Power at Which Flow Instability Is Calculated to Occur in the LEU Core.

Accident Analyses

Accident analyses were completed for a number of hypothetical initiating events, which include rapid and slow reactivity insertions from postulated ejection of experiments and from runaway motion of control rods at their maximum allowed rate, loss-of-flow transients due to pump coast-down and pump seizure, criticality analysis for accidental drop of a fresh fuel assembly onto a spent fuel storage crate, a Design Basis Accident (DBA) involving postulated release of radioactivity, and a Beyond Design Basis Accident (BDBA) in which water is lost from the reactor pool until the core is completely uncovered.

Reactivity Insertion Transients

The OLC allow a maximum reactivity worth of 1.2% Δk/k for all moveable and non-secured experiments in the core. A rapid insertion of this reactivity is the limiting transient for all of the reactivity insertion transients considered. This transient was analyzed using the PARET/ANL code[7]. For a rapid insertion of 1.2% Δk/k with forced convection flow, the reactor period trip was assumed to fail, but the reactor protection system initiated a reactor scram based on the power level safety system setting. The maximum temperatures of the fuel and cladding were calculated to be 164 °C, which is far below the safety limit of 530 °C. Reactor power and peak cladding temperature profiles are shown in Figure 4.

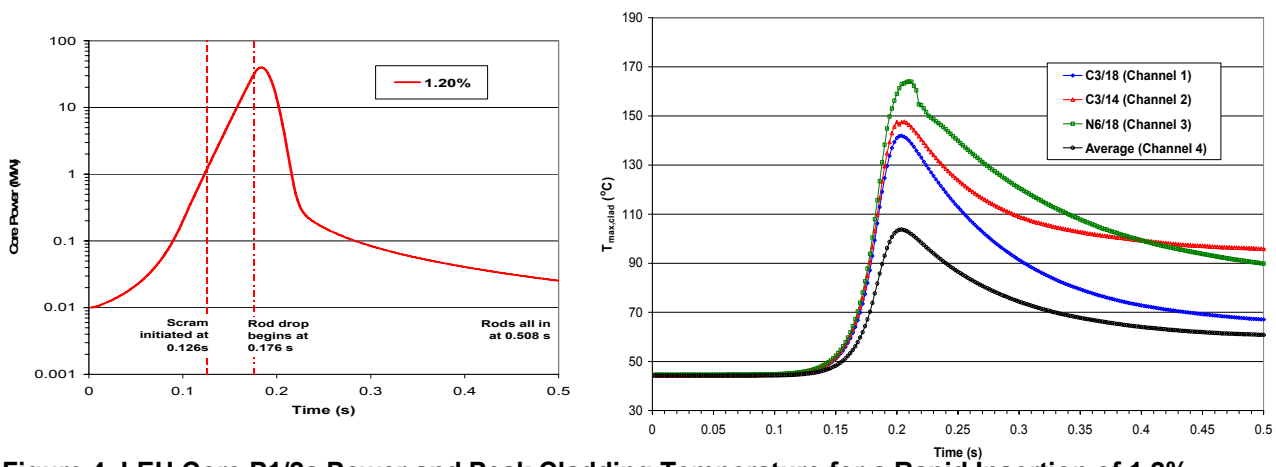


Figure 4. LEU Core P1/2a Power and Peak Cladding Temperature for a Rapid Insertion of 1.2% Δk/k Reactivity from Low Power.

For natural convection cooling, the most limiting reactivity insertion transient is also a rapid reactivity insertion of 1.2% Δk/k. The estimated increase in cladding temperature would be less that 12 °C, resulting in a maximum cladding temperature of about 66 °C. This value is far below the safety limit of 530 °C.

Thus, the SSS that were selected both for the force convection mode of operation and the natural convection mode of operation 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.

Loss-of-Flow Transients Due to Pump Coast-Down and Pump Seizure

Pump coast-down and pump seizure transients were analyzed for the LEU core using the RELAP5-3D code[8], including cases with proper operation of the flapper valve and cases in which the flapper valve does not open. All transients started from nominal power and flow and included a scram on the second scram signal. The pump coast-down curve is based on measurements at ITN. The flow was assumed to drop linearly to zero in 0.1 second for the pump seizure case. The results for a pump coast-down and a pump seizure are shown in Figure 5.

Peak cladding surface temperatures of about 93 ^{o}C were reached before the scram was initiated. After a rapid drop in temperature due to the scram, a peak cladding temperature of about 75 ^{o}C was reached for the pump coast-down case in which the flapper valve failed to open. Similarly, a maximum cladding surface temperature of about 92 ^{o}C was reached for the pump seizure case in which the flapper valve opens. The peak temperatures reached in these transients are far below the safety limit of 530 ^{o}C for the fuel and cladding temperatures.

The LEU core can easily survive loss-of-flow transients due to pump coast-down and pump seizure without any transient coolant temperatures approaching the boiling point. Even if the flapper valve fails to open, natural circulation will be adequate to remove the decay heat.

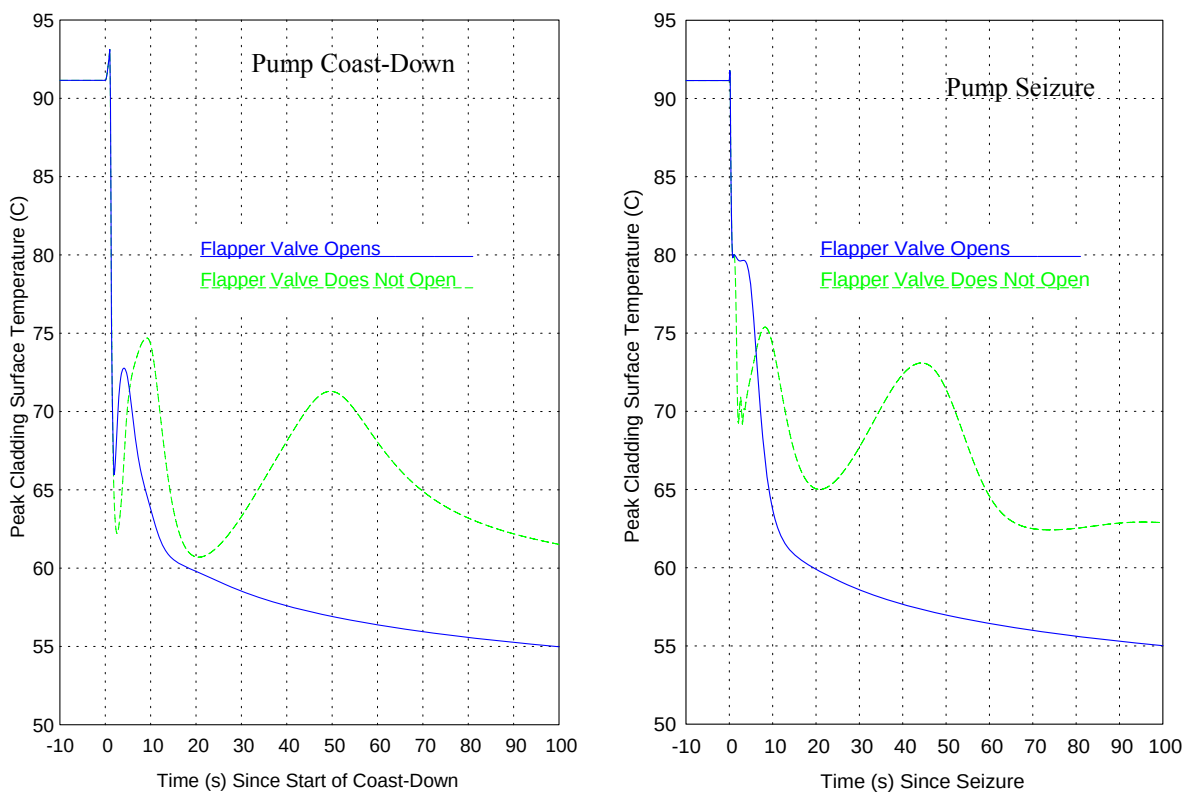


Figure 5. Loss-of-Flow Transients for Pump Coastdown and Pump Seizure for the LEU Core.

Criticality Analysis for Drop of a Fresh Fuel Assembly

The RPI has two aluminium crates that can store up to 20 fuel assemblies each, on a 5x4 matrix inside the reactor pool. No neutron absorbers are used. The nominal distance between the centers of any two cells containing a fuel assembly is 20 cm.

Several cases were analyzed using MCNP. Twenty fresh standard assemblies were considered as a worst case (i.e., no credit was taken for burnup). The accident scenario assumed that an operator dropped one fresh assembly during transport onto a full crate of fresh assemblies that did not have the protection grid installed, as shown in Figure 6. The effective multiplication factor of this configuration was calculated to be 0.53, a value far below critical.

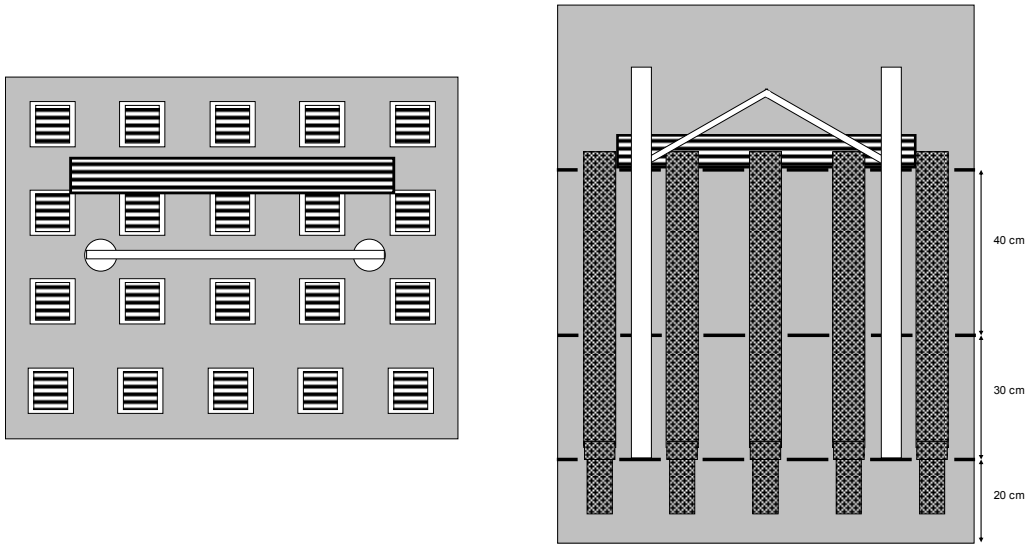


Figure 6. MCNP Layout for Worst-Case Fuel Assembly Drop onto an RPI Storage Crate.

Design Basis Accident

As the Design Basis Accident for the RPI, it is postulated that by some means one or more of the fuel plates in the control element with the highest burnup (C3) undergo partial melting while in the reactor pool. It is assumed that melting affects the equivalent of one fuel plate. A fraction of the fuel plate fission product inventory is released into the pool water and from there to the air of the reactor hall. In a further step, a fraction of the total fission product content of the air in the reactor hall is released to the environment.

The fission product inventory of one fuel plate in the control assembly with highest burnup was calculated using the ORIGEN 2.2 code[9] and a source term was determined taking into account fractional transfers of radioactive isotopes from the melted fuel to the pool water and then from the pool water to the air of the reactor building. The effective (whole body) and thyroid doses[*] were calculated using the COSYMA code[10] for a ground release for the last staff member leaving

[*] Committed Dose Equivalents (CDE), which are calculated for individual organs and tissues over a 50-year period after inhalation.

the reactor hall (exposure time – 5 minutes before evacuation), the maximum exposed member of the public (standing at the campus fence 75 m from RPI for 1.5 hours before evacuation) using a building leakage rate of 20% per hour, and the maximum exposed permanent resident (living 140 m from the RPI site for one year), also using a building leak rate of 20% per hour. For the dose calculations, contributions arising from ingestion of contaminated food were omitted because no professional agricultural activity is present in the area surrounding the RPI. The surroundings consist mainly of industrial, commercial, and urban residential areas. All food consumed in these areas originates from regions that will not be affected by the discussed accident scenario.

The calculated doses for the three key exposed individuals shown in Table 4 are far below the dose limits specified by European Council Directive[11].

Table 4. Comparison of Calculated Doses* and Dose Limits* Specified by European Council Directive for Three Key Exposed Individuals.

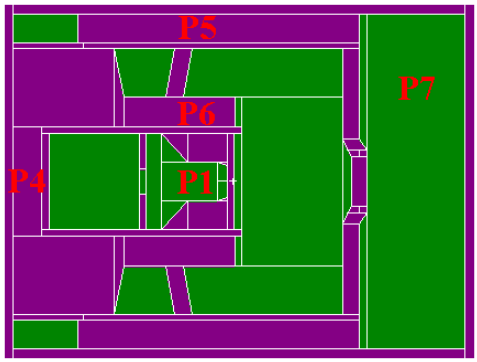
		Effective (Whole Body) Dose, mSv		Thyroid Dose, mSv	
Exposed Individual	Exposure Time	Calculated Dose, mSv	Dose Limit, mSv	Calculated Dose, mSv	Dose Limit, mSv
Maximum Exposed Worker	5 minutes	0.00045	50 / year	0.0061	1000 / year
Maximum Exposed Member of Public	1.5 hours	0.0077	1 / year	0.182	20 / year
Maximum Exposed Permanent Resident	1 year	0.026	1 / year	0.287	20 / year

* Doses and Dose Limits are committed dose equivalents (CDE), which are calculated for individual organs and tissues over a 50-year period after inhalation.

Beyond Design Basis Accident

In this hypothetical accident, water is lost from the reactor pool, due to sudden rupture of the 12 inch diameter pipe of the primary circuit through which water is returned to the pool, until the core is completely uncovered. The maximum cladding temperature was calculated to be about 211°C in the HEU core and 212 ^{o}C in the LEU core using a decay heat curve calculated using ORIGEN 2.2, power distributions calculated using MCNP, and a code developed by ITN based on the methods utilized for analysing a loss-of-coolant accident in Ref. 12. Since this temperature is far below the safety limit of 530 ^{o}C, there would be no melting of the cladding and no release of radioactivity into the air of the reactor hall.

Further, a three dimensional Monte Carlo model of the reactor hall was constructed, including the core and all existing shielding. A photon source term was calculated using ORIGEN 2.2 at shutdown after the reactor had been operated continuously at a power of 1 MW for 500 MWd (the maximum burnup considered for this core) and at eight hours after shutdown. Photon dose rates were then calculated using MCNP at nine key locations in the reactor hall. An example showing five of these locations is shown in Figure 7. The corresponding photon dose rates are shown in Table 5.



P1 is at level 3 of the reactor hall (top of pool) in direct line over the core.

P4 is at level 3 of the reactor hall, near the West wall.

P5 is at level 3 of the reactor hall, near the North wall.

P6 is at level 3 of the reactor hall, North of the pool border.

P7 is at level 3 of the reactor hall, in the control room (attenuation of gamma radiation by the glass windows was neglected).

Figure 7. Horizontal Cut through the Reactor Hall at Level 3 with Some of the Positions for which MCNP Calculations Were Performed.

Table 5. Photon Dose Rates for All Nine Positions at Shutdown and 8 Hours after Shutdown

Condition	Photon dose rate (mSv/h)								
	Position 1	Position 2	Position 3	Position 4	Position 5	Position 6	Position 7	Position 8	Position 9
at shutdown	98100	19.9	65.3	14100	148	221	84.6	50.4	0.00955
shutdown + 8h	18600	4.37	14.4	2400	32.0	46.5	26.0	11.6	0.00294

P2 is at level 1 of the reactor hall (floor level), at the horizontal access to the thermal column.
P3 is at level 2 of the reactor hall (intermediate level), close to the personnel airlock.
P8 is at level 2 of the reactor hall, close to the heavy equipment access door.
P9 is at level 1 of the reactor hall, at the entrance to the pump room.

Positions P3 and P8 cover the two access ways to the reactor hall, through the personnel airlock and heavy equipment access door, respectively. Position P2 covers the access to the floor level of the hall; P9 covers the location of the external pool isolation valves and the access to the pump room. P1, P4, P5 and P6 cover several points in the vicinity of the pool top.

The values in Table 5 show that intervention for this scenario is possible with a limited exposure. The main goal should be to reduce the leakage, to the extent possible, and put water back into the pool. The reduction of the leakage has to be addressed at the floor level or in the pump room, where doses are the lowest. The replenishment of the pool can be addressed in several ways, from inside and outside the reactor building.

Operating Limits and Conditions

Several important changes were made to the Operating Limits and Conditions. The key changes involve the specification of the Safety Limit applicable to both the forced convection and the natural convection modes of operation based on a maximum temperature in the fuel and cladding of 530 ^{o}C and modification of the Safety System Settings to be consistent with the safety analyses that were performed. A change was also made to the maximum allowed reactivity for a moveable experiment based on measurements that were made at ITN. A shutdown margin condition was removed because it was no longer applicable to RPI. Several other conditions were removed because they were superseded by the much more detailed and sophisticated thermal-hydraulic analyses that were performed in this study.

Conclusion

The results of neutronic studies, steady-state thermal-hydraulic analyses, accident analyses, and revisions to the Operating Limits and Conditions demonstrate that the RPI reactor can be operated safely with the new LEU fuel assemblies.

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