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SUBJECT: Forwards responses to NRC 880121 questions re application to use low enriched U fuel.

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February 10, 1988

Mr. Theodore S. Michaels, Project Manager
Standardization and Non-Power Reactor Project Directorate
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Michaels:

Enclosed please find the responses to the questions submitted on January 21, 1988, as a result of our application to use low enriched uranium fuel.

If you require further information, please contact me.

Sincerely,

Thomas H. Newton, Jr.

Thomas H. Newton, Jr.
Director
Nuclear Reactor Facility

THN/ns
encs.

cc: J. A. Mayer Jr.

A020
1/1

Attachment 1   9/10/87 amendment

1. Shutdown Margin

   a) The calculated shutdown reactivity worth with the most reactive control blade (blade no. 3) fully withdrawn and the other control blades fully inserted is 6.5%$\Delta k/k$ for either HEU or LEU fuel loaded in the standard 24 element configuration with the regulating blade fully withdrawn. With the maximum allowed excess reactivity including all experiments of 0.5% $\Delta k/k$, the reactor would be subcritical by 6.0% $\Delta k/k$.

   b) provided in revision of Section 3.1.

2. Control Blade Worths

   a) Section 4.5.2 revised.

3. a) Regulating Blade Worth Section of 2.1 deleted, Section 4.5.3 revised. Compliance with the 0.7% reactivity requirement will be demonstrated through semiannual measurements of regulating blade reactivity worth, concurrent with excess reactivity measurements.

   b. Section 3.1 revised to include surveillance interval.

4. Section 4.4 revised.

Attachment 2   HEU to LEU Conversion

1. Corrected

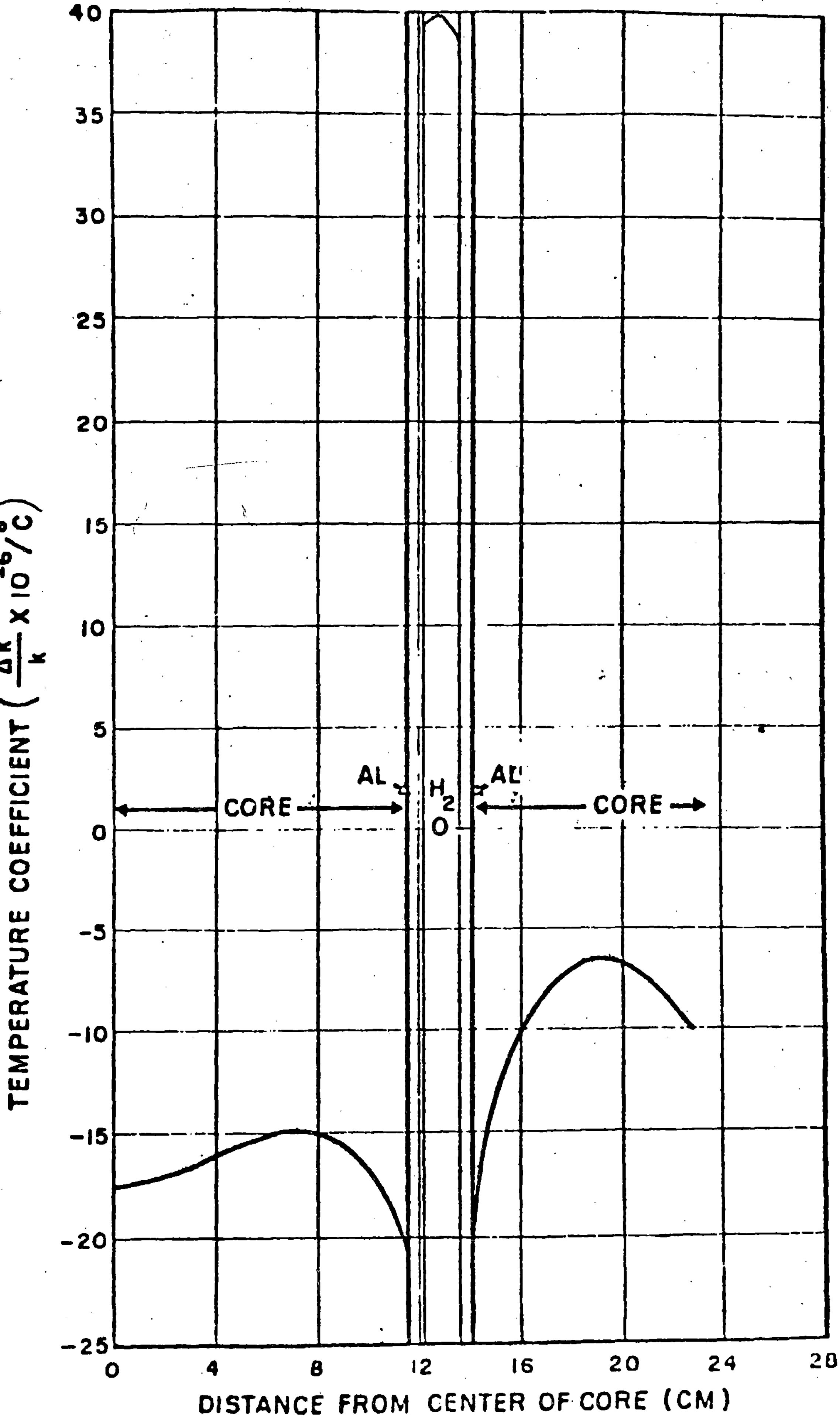
2. Section 5.1 revised to include comparison of HEU parameters of ANL calculations with WPI data.

3. In any power excursion, water surrounding the fuel will be heated to a higher temperature (and lower density). Since temperature and void coefficients are more negative in the fuel regions than in the non-fueled regions, such as shown in Figures 9 and 10 from our current SAR, enclosed, any reactivity changes due to temperature and density changes in the fuel regions would be more negative in an excursion than the stated reactivity coefficients would imply, since they are based on an average change throughout the core region, as is discussed in Section 5.7 of the ANL report.

4. Section 7.3: As was discussed in the 17 September correspondence, the only change was the fuel heat capacity given in Section 7.3.1 was increased slightly (about 2%) for the LEU fuel. The potential consequences of this change is negligible, and if anything, decreases the likelihood of fuel melting. All other parameters discussed in Section 7.3 remain unaffected by the fuel change. The multiplication factor and shutdown reactivity were changed in response to questions 11, but

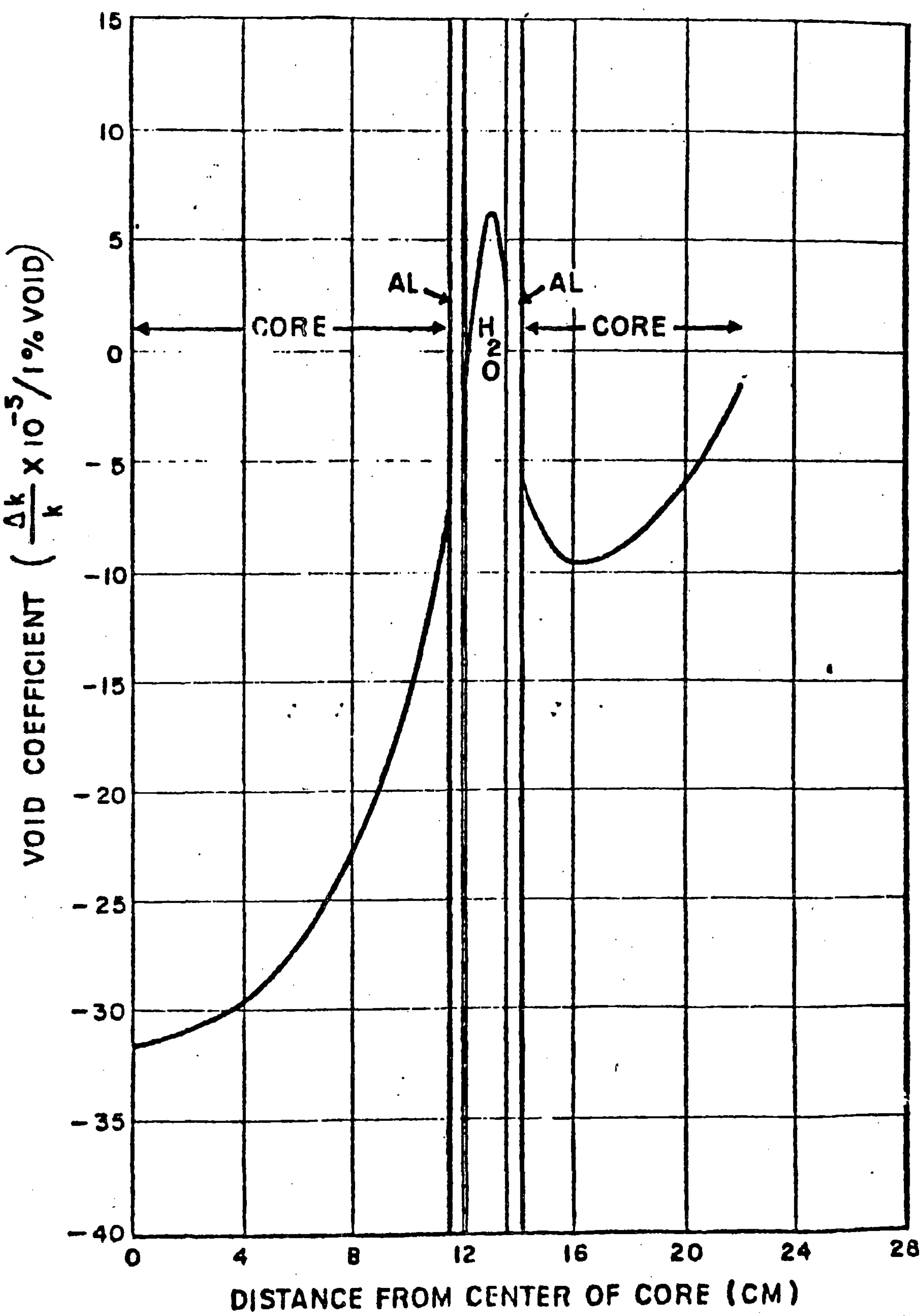
Figure

TEMPERATURE COEFFICIENT



1-27-82

Figure 10
VOID COEFFICIENT



-27-82

have little effect on the order of magnitude estimation discussed.

Section 7.4: The only change occurred in Section 7.4.2, p. 44, which involves the release of fission products from one fuel plate. The calculations were changed because of the change in dimensions to the 18 plate LEU fuel elements. All accidents discussed in Section 7.4 are not significantly changed with the conversion to LEU including the fission product release consequences discussed in Section 7.4.2. The only difference in fission product relase, as discussed in Section 6.2 of the ANL report, is the presence of very small quantities of plutonium, which has a negligible effect on radiological consequences.

5. Since the LEU prompt neutron lifetime is only about 10% lower than the HEU lifetime, the conversion to LEU would have little bearing on the results given in the postulated startup accident in Section. 7.3.1, particularly in the order-of- magnitude estimations of weak source conditions where statistical fluctuations predominate.

6. Each specific parametric change and its effect on reactor operation was discussed in the September 17, 1987, correspondence. As was discussed in the correspondence, most of the changes had a negligible or conservative effect on operations. Only those parameters which significantly affect reactor operation will be reiterated below.

Section 2.3: A column listing ANL calculated values for the HEU core was added. In four cases the ANL values for the HEU core differed from the original GE calculations. In the three cases of prompt neutron lifetime, temperature coefficient, and power peaking - hot channel factors, insufficient documentation exists to determine the values and assumptions made in the G.E. calculations performed in 1959. In the case of neutron fluxes, different energy groups were used, and the values reflect this difference.

Power peaking factors - the maximum power peaking factor is 9% higher in the LEU core. This difference is small and should not significantly affect the thermal-hydraulic operation of the reactor, particularly with the engineering hot channel factor of 1.51.

Temperature and void coefficients - These reactivity coefficients show a slight change to the conservative side, indicating that for any increase in temperature or voiding, the reactivity change will be slightly more negative than in the HEU core.

Prompt neutron lifetime - the neutron lifetime in the LEU core is 12% smaller than that of the HEU core. This indicates that for a given reactivity change, the period will be comparably smaller in the LEU core.

Section 4.1: Only fuel element dimensions changed in Section 4.1.1.

Section 5.2.2: A column listing ANL calculated values for the HEU neutron fluxes was added. The comparative values reflect the hardening of the neutron spectrum with the LEU fuel.

Section 5.2.6: The HEU prompt neutron lifetime was listed along with the LEU value. The shorter LEU lifetime will result in only a 12% smaller period than the HEU core with the same reactivity addition, as discussed above.

Section 5.2.7: Reactivity changes with core configuration changes are shown with HEU values added. These indicate very little, if any, differences between the two cores.

7. Changes made to Section 2.3.

8. As discussed in Section 4.3.10 of the WPI SAR, the blade to be moved is selected by a three position switch and then is withdrawn or inserted by a pistol-grip switch with a spring return. It is not possible to select more than one control blade to be moved at one time. There is also an interlock to prevent withdrawal of a control blade with the regulating blade withdrawn.

9. Section 5.2.5 title corrected.

10. The Borax I reactor had a core consisting of aluminum clad, aluminide HEU fuel in fuel plates, which is similar to the fuel used in the WPI reactor. Since the calculated values of neutron lifetime worth for the LEU and HEU fuel in the WPI reactor are comparable to the measured Borax I values, comparison of the effects of reactivity insertions between the reactors is useful, as is indicated in the expanded second sentence of Section 5.2.6.

11. Shutdown reactivity changed to 12%, along with the resulting change in multiplication factors to 9. This has little effect on the order-of-magnitude estimation discussed. The discussion assumes that criticality ocurs during maximum withdrawal of a safety blade at 7.5 in/min, with a differential reactivity worth of about 0.24% $\Delta k/k$/inch.

12. Our current fuel storage capabilities consist of three underwater fuel storage racks with a capacity of 9 elements each, as well as five fuel storage boxes for storage of low activity fuel out of the reactor pool. When the fuel change is to take place, the HEU elements will be completely unloaded from the core into the fuel storage racks, leaving 1 to 3 spaces in the racks available for interim storage of fuel, depending upon the number of HEU fuel elements removed from the core. When it becomes appropriate to load or add LEU fuel to the reactor during subcritical testing, each LEU fuel element will be removed from its shipping container, placed in an available space in the fuel storage racks, and then placed in the designated reactor position. With the minimum anticipated core loading of 24 LEU elements, sufficient space exists in the fuel storage racks and boxes to store the remaining 4 LEU elements. If it becomes necessary to completely unload the LEU core after operation, the HEU fuel will be removed to appropriate containers, such as a DOT-6M shipping container and monitored for radiation level and security intrusion.

13. The removable plate elements are used as fuel in approach to critical measurements and to provide increased reactivity where needed in different core configurations. They are not experiments as defined in Section 1.0 of the technical specifications and are never used with a movable component. The limitation on use is the same for the elements whether treated as fuel or experiments: the maximum excess reactivity of 0.5% $\Delta k/k$. The reference to the removable plate elements was removed from Section 2.4.

2.0 SAFETY LIMITS AND OPERATING RESTRICTIONS

2.1 Safety Limits

Criticality: The reactor shall be subcritical when the three control blades are at their fully withdrawn positions and the regulating blade is in its fully inserted position.

Shutdown Margin: The minimum shutdown margin under any condition with the highest worth control blade fully withdrawn shall be no less than 1% $\Delta k/k$.

Magnet Release and Blade Drop Times: The interval between the occurrence of cutoff voltage (scram) and the separation of each control blade from its magnet shall not exceed 100 msec. Total time of insertion of the first 24 in. of the control blades following initiation of a scram signal shall be less than 600 msec, including the magnet release time.

Maximum Excess Reactivity: The maximum excess reactivity above cold, clean, critical shall be 0.5% $\Delta k/k$.

Radiation Alarms: Upon indication of radiation levels in excess of 50 mrems/hr (20 mrems/hr for fuel storage) area monitors shall actuate audible evacuation alarms in the reactor room and in the second and third floor areas above the reactor pool.

Radiation Levels: The maximum radiation levels 1 m above the pool surface and at the surface of the concrete shield, when the beam port and thermal column are closed, shall be less than 50 mrems/hr.

Control Blade Withdrawal: The maximum withdrawal rate for a control blade shall be 7.5 in./min. The maximum reactivity addition rate through movement of the regulating blade shall be 0.006% $\Delta k/k$ sec. Interlocks shall prevent simultaneous withdrawal of more than one control blade and shall prevent withdrawal of any control blade unless the regulating blade is fully inserted.

Startup Source Requirement: During reactor startup, a neutron source producing at least 10 neutrons/sec shall be located adjacent to the fuel region. When readings on the log count rate meter are below 50 counts/sec, an interlock shall prevent withdrawal of any control blade.

Temperature and Void Coefficients: The temperature and void coefficients of reactivity shall be more negative than $-2 \times 10^{-5} \Delta k/k\ ^{o}F$ and -2×10^{-3} $\Delta k/k$ % void, respectively, at $80^{o}F$.

Water Level: The minimum depth of water above the top of the end box of the core fuel elements in the reactor pool shall be 10 ft.

Water Purity: Corrective action shall be taken promptly if the following limits for the pool water are not met:

3.0 SURVEILLANCE REQUIREMENTS

3.1 Frequency of Surveillance

Daily: Before each day's critical operation (with the exception of those experiments that require the reactor to be operated continuously for more than one full day), the two safety channels, the log-N period channel, and the console annunciator system shall be checked and ensured to be operational.

Quarterly: The area radiation monitoring systems and the pool water level switch shall be checked and ensured to be operational quarterly.

Senuannually: At least semiannually, a reactor inspection shall be performed consisting of

(1) The excess reactivity of the core above cold, clean, critical shall be measured.
(2) The console instrumentation shall be calibrated by a foil activation measurement of reactor power where applicable, or calibrated by other means, and checked for proper conditions.
(3) Pool water pH shall be measured and conductivity and pH devices shall be calibrated.
(4) The minimum shutdown margin with the highest worth control blade fully withdrawn shall be verified to be no less than 1% $\Delta k/k$.
(5) The reactivity worth of the regulating blade shall be measured.

Annually: At least once each year, all fuel elements shall be removed from the storage racks. While the fuel elements are thus stored, the control blades shall be brought to the surface and visually inspected and the blade drives lubricated. Blade drop times and magnet release times shall be measured for each control blade, and a plot of blade drop times versus distance shall be obtained for each safety blade and compared with data of previous years. Abnormal deviation from previous data will be investigated and reviewed by the Radiation, Health, and Safeguards Committee.

3.2 Action to be Taken

If maintenance or recalibration is required for any of the items, it shall be performed and the instrument shall be rechecked before reactor startup proceeds.

3.3 Radiation Detection

Area Monitors: Area radiation sensors capable of detecting gamma radiation in the range of 0.1 to 100 mrems/hr shall be installed near the beam port, demineralizer, thermal column door, fuel storage area, and less than 1 m above the core pool surface. Upon indication of radiation levels in excess of 50 mrems/hr (20 mrems/hr for fuel storage) these monitors shall actuate audible alarms in the reactor room and in the second and third floor areas above the reactor pool.

4.0 SITE AND DESIGN FEATURES

4.1 Site

The reactor and associated equipment is housed in the Washburn Laboratories located between West Street and Boynton Street on the campus of Worcester Polytechnic Institute in Worcester, Massachusetts.

4.2 Restricted Area and Exclusion Area

The reactor room shall constitute a restricted area as defined in 10 CFR 20 and shall be controlled by partitions and normally locked doors. In addition, two small areas, one each on the second and third floors of Washburn Laboratories, directly above the reactor control drives, shall become restricted areas whenever the reactor is operating at power levels in excess in 1 kW and radiation levels in any of the rooms exceed 2 mrem/hr. The exclusion areas, as defined in 10 CFR 100, shall consist of the reactor room and the areas above the reactor.

4.3 Reactor Building and Ventilation System

The reactor shall be housed in a closed room that is designed to restrict leakage. The ventilation system shall provide at least two changes of air per hour in the reactor room whenever the reactor is operating.

4.4 Reactor Core

Fuel Elements: Standard fuel elements shall be flat plate type consisting of uranium-aluminum alloy clad with aluminum. The width and depth of each fuel element shall be 3 in. x 3 in. Each element shall have an active length of 24 in. There shall be a maximum of 10 g of U-235 in each fuel plate and not more than 170 g of U-235 in any fuel element. The fuel shall be enriched to less than 20% U-235. Standard fuel elements have 18 fuel plates, each plate 1.52 mm thick with a clad thickness of 0.381 mm on each side. A maximum of 28 standard fuel elements may be installed in the core.

Not more than two experimental fuel elements with sixteen removable fuel plates similar to standard fuel plates and fitted with removable top end boxes may be installed in the core. These elements may be used as part of the core assembly either as complete elements or as partial assemblies, loaded with from 2 to a total of 18 fuel plates each.

4.5 Reactor Safety and Control Systems

The safety system shall be designed so that no single electrical fault that partially or completely disables the automatic scram function can, in any manner, impair or disable the manual scram function, and vice versa. The safety system shall be fail safe with respect to loss of voltage.

4.5.1 Nuclear Instrumentation

The channels of nucler instrumentation (listed below with their minimum operating ranges) shall during all reactor critical operations be operational and shall be connected to the safety system, except as noted in Table 4.1.

(1) startup channel, background to 10^{-2}% full power, i.e., background to 1 W

(2) log-N period channel; 2 x 10^{-3}% to 150% full power; i.e., 0.2 W to 15 kW

(3) Linear safety channels 1 and 2; 2 x 10^{-3}% to 150% full power; i.e., 0.2 W to 15 kW

4.5.2 Control Blades

There shall be three control blades intersecting the core, each consisting of vertical blades 10.5 in. wide x 40.5 in. long with a poison section composed of boron carbide and aluminum 0.375 in. thick sandwiched between aluminum side plates.

4.5.3 Regulating Blade

There shall be one regulating blade consisting of a vertical stainless-steel blade 10.65 in. wide x 40.5 in. long x 0.125 in. thick. It shall have a reactivity worth of less than 0.7% $\Delta k/k$.

4.5.4 Blade Position Indicators

The blade position indicator on the console shall provide an indication of the blade position to within $\pm$ .02 in. Signal lights shall be provided for each control blade drive and for the regulating blade to indicate the upper and lower limits of travel and, in the case of control blades, an armature engaged by a magnet.

2.3 Summary of Calculated Reactor Data (Furnished by Argonne National Laboratory (1)

Tabulated below are the significant design parameters for both the High Enriched and Low Enriched Fuels used in this reactor.

	LEU	HEU
Reactor Materials		
Fuel	Uranium aluminum alloy, 19.75% enriched	UAl 93% enriched
Moderator	High purity light water	
Reflector	High purity light water and graphite	
Coolant	High purity light water	
Control	Boral and stainless steel	
Structural material	Aluminum	
Shield	Water and aluminum lined concrete	
Structural Dimensions:		
Pool	8x8 by 15 ft. deep	
Core (active portion)	15x15 by 24 inches high	
Grid box	9x6 array of 3 inch modules	
Beam port	One, 6 inch diameter	
Thermal column	One, 40x40 inches in cross-section	
Strategic Materials:		
Fissionable material	4.2 Kg U-235	3.4 Kg U-235
Burn-up	Approximately 1% U-235	
Fuel life	Limited by factors other than burn-up	
Thermal Characteristics: (Calc)		
Operating power	10 kw (maximum)	
Temperature, water	130 deg. F (maximum)	
Maximum Power peaking factor	2.17	1.99
Maximum hot channel factor	1.51	1.51
Maximum heat flux	400 Btu/hr.-sq.ft.	
Specific power (clean,cold)	2.5 watt/gm U-235	3.0 watt/g U-235
Maximum gamma heat in core	11 watt/liter	10 watt/L
Nuclear Characteristics: (Calc)		
Average thermal flux	8.3×10^{10} n/Cm^{2}sec nv	9×10^{10} nv
Average fast flux	11×10^{10} n/Cm^{2}sec nv	11×10^{10} nv
Maximum operating excess reactivity	0.5% Δkeff.	0.5% Δkeff

	LEU	HEU
Nuclear Characteristics (Cont.)		
Critical mass	4.0 Kg U-235	3.3 Kg U-235
Temperature Coefficient	-1.63×10^{-4} Δkeff. per degree C	-1.7×10^{-4} $\Delta k/^{o}C$
Void coefficient	-2.4×10^{-3} Δkeff. per 1% void	-2.0×10^{-3} $\Delta K/\%$
Prompt neutron lifetime	61.2 μseconds	69.0 μsec
Delayed neutron fraction	0.0077	0.0077
Control		
Safety Elements		
Number	3 vertical blades	
Dimensions	10.5 inches wide by 40.5 in. long by 0.375 in. thick	
Material	Boral	
Reactivity control	3.5% each, minimum	
Total worth, 3 blades	12% $\Delta k_{eff}/k$	
Maximum withdrawal rate	7.5 inch/min.	
Regulating Element		
Number	1 vertical blade	
Dimensions	10.65 in. wide by 40.5 in. long by 0.125 in. thick	
Material	Stainless steel	
Reactivity control	0.7% $\Delta k_{eff}/k$	
Maximum withdrawal rate	3.8 inch/min.	
Standard Fuel		
Type	Flat plate	
Number of elements	24 for minimum critical loading	
Fuel alloy	Uranium-aluminum	
Clad material	Aluminum	
Fuel enrichment	U-235 19.7% enriched	93% enriched
Number of plates per element	18	10
Plate thickness	1.52 mm (0.060 in.)	2.51 mm (.099 in.)
Cooling System		
Coolant	Pool water	
Type cooling	Natural convection	
Temperature	130 deg.F (maximum)	
Purity required	1 ppm - 5×10^{5} Ohm-cm	

2.4 Experimental Facilities

The experimental facilities provided with the reactor are a thermal column, and a 6 inch beam tube.

REFERENCES

1. J.E. Matos and K.E. Freese, "Analyses for Conversion of the Worcester Polytechnic Institute Reactor from HEU to LEU Fuel," RERTR Program, Argonne National Laboratory, Argonne, IL, August 1987.

SECTION 5

REACTOR OPERATING CHARACTERISTICS

5.1 Introduction

This section summarizes reactor operating characteristics as calculated by the General Electric Company in support of the initial reactor designs and by Argonne National Laboratory in support of the fuel change to low enriched uranium (LEU) (1). The ANL calculations were also performed for the WPI Reactor with highly enriched uranium (HEU) and checked against existing WPI data for HEU fuel. The G.E. calculations were "checked, where possible against data from the Bulk Shielding Facility, the Geneva Swimming Pool Reactor, and critical experiments at ORNL" and were confirmed by "low-power tests and actual operation of the Spanish Pool Reactor," modified where necessary to reflect the increase from the original 1 kw to 10 kw in 1967.

5.2 Nuclear Characteristics

The critical mass obtained from ANL diffusion calculations is 4.0 kg of U-235 in 20% enriched fuel for the 5-by-5 element core surrounded by water. The initial core loading consisted of 3.50 kg of U-235 in highly enriched fuel with an excess reactivity of 0.23% Δk/k.

5.2.2 Neutron and Gamma Flux

Average neutron fluxes for the reactor core at 10 kw are as follows:

	LEU	HEU
thermal (<.625 ev):	8.3×10^{10} nv	9.6×10^{10} nv
epi-thermal (≥.625 ev):	5.5×10^{10} nv	5.4×10^{10} nv
fast (>5.53 Kev):	11.3×10^{10} nv	11.1×10^{10} nv

The distribution of the midplane fluxes in the core is shown in figure 6.

The gamma heating in the core has a peak value of about 11 watt/liter. Figure 7 shows the distribution of the midplane thermal, epithermal, and fast fluxes in the thermal column. Dose rates at 10 kw in the thermal column are as follows:

gamma dose at shield face: 3 rem/hr

gamma dose at outside of thermal column door: <10 mrem/hr

maximum neutron dose at outside of thermal column door from streaming: <10 mrem/hr

5.2.3 Control System

The three safety blades, Figure 13, cut the core into three sections. The control blade with maximum reactivity is blade no 3. When this blade is fully withdrawn, control blade no. 1 and 2 have a combined reactivity worth of 6.5% $\Delta K/K$. The control blades span almost the full width of the core. As a result, they keep the reactor subcritical even if a loading error were made and more fuel included than planned. To go critical with the entire 6 x 9 grid filled with fuel elements would require at least 300 gm of U-235 per element, about twice the loading of the fuel elements intended to be used.

5.2.4 Burn-up

The burn-up is estimated to be approximately 1% U-235. The fuel life will be limited by factors other than burn-up.

5.2.5 Temperature, Doppler and Void Coefficients

Negative temperature and void coefficients aid stable operation of the reactor. The coefficients have been computed separately as functions of temperature or void fraction for each of three physical effects: (1) hardening of the neutron spectrum due to an increase in the water temperature only, (2) the increase in neutron leakage due a decrease in the water density only, and (3) the increase in absorption of the ^{238}U epithermal resonances due to an increase of the temperature of the fuel meat only.

Slope of the core average reactivity feedback coefficients between 25°C and 30°C, along with the whole core void coefficient for a 1% change in water density only are compared with those of the WPI core with 93% enriched fuel below.

Core Average Temperature, Void, and Dopper
Coefficients With All Blades Withdrawn

Effect		HEU Core	LEU Core
Water Temperature Only		-1.7	-1.63
Water Density Only,		-0.5	-0.6
$\Delta k/k \times 10^{-4}$ per °C	SUM	-2.2	-2.2
Fuel Temperature (Doppler) Only, $\Delta k/k \times 10^{-5}$ per °C		0.0	-1.8
Void Coefficient (0-1% Void), $\Delta k/k \times 10^{-3}$ per % void		-2.0	-2.4

The effect of water temperature change on reactivity is shown in figure 8. With the water density held constant, the temperature at which the maximum loaded excess reactivity would be offset is about 50°C. Figure 9 shows the effect of water density change on reactivity of the core. In an excursion, the water outside the core and in any control blade gaps would not be heated to the same temperature and density as the water

flowing through the fuel elements. Therefore, the values of the reactivity coefficients shown in Section 2.3 are reasonably conservative.

The effect of fuel temperature change on reactivity due to Duppler broadening is shown in figure 10. Reactivity effects of fuel temperature, water density, and water temperature are all additive so that any credible postulated excursion would be easily offset before temperatures could approach that of fuel damage.

5.2.6 Neutron Lifetime

The prompt neutron lifetime has been calculated as 6.1×10^{-5} seconds for the LEU core and 6.9×10^{-5} seconds for the HEU core. These values are comparable to the measured value of 6.5×10^{-5} on the Borax I reactor, and indicate that for the same amount of reactivity inserted in the WPI LEU core, a slightly shorter period will result than in the Borax I.

5.2.7 Alteration of Core Geometry

Alterations of core geometry will affect the available reactivity. If the reactor is loaded to criticality with a square array of 25 fuel elements, arranged symmetrically with respect to the control blades (figure 13), the changes in reactivity are estimated to be as follows:

		LEU	HEU
1.	An extra fuel element is added	+ 1% Δk_{eff}	+ 1%
2.	A center fuel element is removed	- 4.6% Δk_{eff}	- 5.1%
3.	The 5 elements in Row B are moved to positions (8, D8, E8, C2 and E2)	- 0.8% Δk_{eff}	- 0.6%
4.	The 4 elements in row F are moved to positions A4 - A7	- 1.0% Δk_{eff}	- 1.0%

Adding an extra fuel element will not make the reactor critical provided that at least one safety blade is in the core.

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1. J.E. Matos and K.E. Freese, "Analysis for Conversion of the Worcester Polytechnic Institute Reactor from HEU to LEU Fuel," RERTR Program, Argonne Nat. Lab., Argonne, IL, August 1987.

7.3 Accidents of Operating Type

7.3.1 Startup Accident

Analysis indicates that the fuel elements will not melt if a startup accident should occur. A pessimistic estimate of the energy released in a startup accident has been made under the assumption the log N-period channel fails to operate, and that scram does not take place until the power reached 1-1/2 times normal operating level. The maximum withdrawal speed of the safety blades is 7.5 in/min. The time delay from generation of a scram signal to the instant when the safety blades are free to drop is less than 100 milliseconds. During scram, the blades are assumed to accelerate (for purposes of the accident evaluation) at rates corresponding to the 1-second drop durve in Figure 11. The energy release is about 40 KJ, giving a peak temperatyre rise of less than 1 degree C. Fuel heat capabity is 2.35×10^6 J/m^{3}K.

The calculation assumes the neutron source to be in place and uses a conservative ratio of trip power to source power, equal to 10^{12}. If the source is absent from the core, this ratio may be of the order of 10^{16} for a clean core. The "source power" then corresponds to the spontaneous fission rate of U^{235} (0.3 fission/sec/kg[1]), and a multiplication factor of 9, due to 12% shutdown reactivity. Startup under this condition is normally prevented by an interlock which requires the B10 startup counter to read at least 5 counts/sec (Section 4.3.10). If however, the interlock were inoperative, and the blades were withdrawn, the final period would be only about 10% shorter[2] than in the case considered above, with unimportant effect upon the fuel temperature rise.

7.3.2 Refueling Accident

Erroneous loading of a fuel element should be easy to avoid because of the small core size, good visibility of the core, and easily recognizable reference points. All fuel loading will be under the supervision of a senior licensed operator and will be in accord with the provisions of Section 6.

It is not normally possible to go critical with the safety blades in the core. In the event a loading error had taken place, the reactor could conceivably go critical with one safety blade partially in the core. Since the position of the blades is indicated on the control console, the operator would now that an error had been made and could shut down the reactor before going to power.

7.3.3 Mishandling the Demineralizer Resin

Following an accident such as cladding failure of a fuel element, the resin would become radioactive and the dose rate at the demineralizer may temporarily rise as high as 10 rem/hr. To avoid such peaks of radioactivity, replacement of the resin, would be scheduled when the dose rate had decayed to a low value. In normal facility operation it is expected that the resin in the demineralizer would be transferred rarely, only two resin changes have been required during the past nineteen years.

With a compartment volume of over 30,000 cubic feet, the concentration of fission products inside would create an exposure rate of the order of 40 rem/hr, giving personnel adequate time to take measures for the protection of the facility and for their own safety. The dose rate outside the building does not constitute a significant hazard to the public. These figures are very conservative in that in actuality the reactor is at power less than 10% of the total hours in a year.

Loss of Coolant with Damaged Fuel

Suppose that a leak develops and the pool drains at a moderate rate, similar to the accident reviewed in Section 7.2.4, but made more serious by the fact that the cladding on one or more fuel elements is already damaged. As before, assume that the reactor had been running at full power up to the beginning of the accident, and that no emergency measures are taken to prevent pool drainage or to cool the core.

To simplify a probably much more complex situation, suppose that a single plate of one element is completely stripped of cladding, and that the fission products released are those which vaporize below 1000 degrees F, i.e. xenon, krypton, bromine, and iodine. Assume that the gaseous fission products are uniformly distributed throughout the core fuel and that all the gas atoms within recoil range (5 microns) of the surface are emitted.(1)

The release of fission products is calculated as follows:

Volume fraction released to building:

$$\frac{1 \text{ element}}{25 \text{ elements}} \times \frac{1 \text{ plate}}{18 \text{ plates}} \times$$

$$\frac{2 \times 5 \text{ microns}}{30 \text{ mils}} \times \frac{1 \text{ mil}}{25.4 \text{ microns}} = 2.9 \times 10^{-5}$$

Total energy release rate by fission products released to building:

9.42 E 9 MeV/sec from γs

3.91 E 9 MeV/sec from βs

With a building volume of over 30,000 cubic feet, the concentration of fission products inside would create an exposure rate of the order of 0.2 rem/hour, with no significant hazard to the public.

Spill of a Radioisotope

If a vial or an isotope solution being irradiated should disperse the isotope in the pool water, it would be removed by the pool cleanup demineralizer. The worse condition of this type is considered to be the spilling of a vial containing $AuCl_3$ in solution after it has been irradiated for five days in a flux of 10^{11} nv. It is calculated that a 10 gram sample dispersed in the pool water would produce 140 mrem/hr at the pool surface and area monitoring equipment will sound an alarm. The room