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HW-69168 SUP

PLUTONIUM RECYCLE CRITICAL FACILITY FINAL SAFEGUARDS ANALYSIS SUPPLEMENTS

F. SWANBERG, EDITOR

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PLUTONIUM RECYCLE CRITICAL FACILITY
FINAL SAFEGUARDS ANALYSIS
SUPPLEMENT

F. Swanberg, Editor

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INTRODUCTION

These supplements to HW-69168, Plutonium Recycle Critical Facility, Final Safeguards Analyses were prepared at the request of the Richland Operations Office of the Atomic Energy Commission to expand the information given in the document and to resolve previously unreviewed safety questions. The broad capabilities and versatility of the basic facility make possible a variety of critical tests with light water and heavy water moderated reactor cores. Several lattice structures and a variety of fissile fuel loadings can be studied.

SUMMARY

Supplement I, "Additional Studies," reports the results of analyses of the nuclear characteristics of heavy water moderated PRCF cores including an all plutonium fuel loading. The consequences of an excursion assuming failure of the safety circuit are presented. Moderator void formation and the negative fuel temperature coefficient of reactivity are shown to be sufficient to arrest the excursion.

Heat transfer analyses of a short-cooled, preirradiated, plutonium-aluminum fuel element are reported assuming the fuel element in air with only convection cooling. Fuel temperatures in excess of the melting point of aluminum but less than the melting point of the Zircaloy-2 cladding of the fuel are indicated. It is postulated that the resultant stress would cause the Zircaloy-2 cladding to fail releasing radionuclides in the reactor cell. The consequences are estimated to be significantly less than those concluded from the study of the maximum credible accident reported in HW-69168.

The consequences of the maximum credible accident, recalculated assuming unfiltered leakage from confinement, are reported. The results

include the estimated quantity of plutonium deposition in off-site persons from inhalation of air-borne radioactive material, radiation exposure from released fission products, and the area in which it may be necessary to confiscate milk and produce.

Supplement II, Analysis of Light Water Moderation, reports the detailed safeguards analyses for critical experiments with light water moderator in the PRCF. Analyses of nuclear excursions with PuO_2-UO_2 fuel to be used in the Experimental Boiling Water Reactor are reported. None of the postulated accidents result in fuel melting and therefore there is no associated release of radionuclides. Limitations are stated which restrict operation of the PRCF to tests with mixed oxide fuels, moderator-to-fuel ratios which give zero or negative void coefficients of reactivity and reactor cores nominally 4 ft. high.

Supplement III, Additional Studies—Light Water Moderation, extends the studies related to light water moderated cores with mixed oxide fuel. The enrichment level assumed is less than previously studied and reported in Supplement II (1.5% rather than 2.5% plutonium). The results of transient analyses with the lower enrichment are presented.

Supplement IV, Analysis of Various Light Water Moderated Core Loadings, reports analyses of critical experiments planned for PRCF when modified for tests supporting the Plutonium Recycle Program. Results are reported for light water moderated plutonium-aluminum fuel loadings with enrichment levels of 1.8 and 5.0% plutonium. Additionally, analyses of PuO_2 fuel loadings with 1.8 wt% PuO_2 are reported. It is assumed that transients or excursions are either terminated by the safety system or terminated by inherent negative reactivity effects when the safety circuit fails. Analog simulations reported indicate no perceptible fuel temperature rise when the transient is terminated by the safety system. When safety circuit failure is assumed, Doppler broadening is the principal overriding effect in ceramic fuel loadings. In metallic plutonium-aluminum fuel loadings, however, moderator boiling at the fuel element surface has almost immediate effect when the moderator void coefficient of reactivity is negative. The magnitude of the initial power peak is of the same order of magnitude for all fuel loadings studied and realtively independent of enrichment levels when safety circuit failure has been assumed. It is concluded that the consequences of credible accidents in light-water moderated reactor configurations would be less severe than the maximum accident reported in Supplement I.

SUPPLEMENT I: ADDITIONAL STUDIES

by

W. K. Winegardner and N. G. Wittenbrock

A. INTRODUCTION

Review, by the Richland Operations Office, U. S. Atomic Energy Commission, of the document HW-69168, Plutonium Recycle Critical Facility, Final Safeguards Analysis,[1] raised a number of questions about the facility. Additional information was requested by the Richland Operations Office in a letter to the General Electric Company which is reproduced as Appendix A of this report.

The additional information presented in this supplement is submitted as the basis for full approval to operate the Plutonium Recycle Critical Facility (PRCF) with a heavy water moderated core, including a variety of core loadings and tests with irradiated fuel elements.

B. SUMMARY

Accident analyses have been completed for various core loadings of the PRCF representing the range of nuclear characteristics that will be encountered in the experimental program for D_2O moderated cores. These analyses showed that all of the core loadings described in HW-69168, including an all plutonium loading, can be operated as safely as the zoned loadings which were analysed in detail in that report. Analyses of the consequences of failure of the safety circuit to trip showed that the effect of the negative fuel temperature coefficient and void formation in the moderator would be strong enough to arrest an excursion, but reactor shutdown following the accident would have to be accomplished by the insertion of poison. This could be done by manually deenergizing the rod magnet circuit.

Heat transfer analyses of a short-cooled Pu-Al fuel element cooled by free convection in air indicated that the equilibrium temperature of the hot-test fuel rod would exceed the melting point of aluminum but would be considerably lower than the melting point of the Zircaloy-2 jacket. However, the probable stress on the Zircaloy-2 jacket and the physical properties of

(1) J. K. Anderson and W. K. Winegardner. Plutonium Recycle Critical Facility, Final Safeguards Analysis, HW-69168. February, 1962.

Zircaloy-2 at the equilibrium temperature for air cooling lead to the conclusion that the molten core of the center rod probably would cause jacket failure. The release to the cell of radionuclides, however, would be significantly less than the release in the maximum credible accident reported in HW-69168.

The radiological consequences of the maximum credible accident were recalculated, assuming 50 ft^3 unfiltered leakage from the confinement cell. The maximum lung uptake of plutonium for offsite persons was calculated to be less than 5×10^{-3} μCi. Although fission product contamination could result in technical overexposure to offsite persons, no detectable biological effects would be expected. Confiscation of milk and produce might be required in an area of 2 mi^2 as compared to the previously reported area of 0.2 mi^2.

C. DISCUSSION

1. Kinetics Analyses of Various Fuel Loadings

a. Planned Fuel Loadings

The experimental program in the D_2O moderated PRCF will be conducted with fuel elements containing cores of UO_2, Pu-Al, UO_2-PuO_2, ThO_2-PuO_2, and those types which will provide a core with similar physics characteristics. As indicated in HW-69168, the fuel will be loaded into the reactor to give zoned loadings (enriched region, buffer region, and test region at the center), quasi-uniform* loadings of enriched fuel elements and natural UO_2 fuel elements, all enriched fuel element loadings, and all natural uranium fuel element loadings. The all natural uranium fuel loadings provide insufficient reactivity for critical experiments.

Experiments will be conducted over a range of critical moderator levels from 5 to 9 ft and, insofar as possible, the available excess reactivity will be limited to 1 \$. The number of fuel elements and the ratio of enriched to natural uranium fuel elements will vary with the planned critical moderator level and the core loading chosen for the experiment. Although a large number

*A quasi-uniform core loading is one in which the UO_2 and Pu-Al fuel elements are distributed as uniformly as possible throughout the core.

of different arrangements of the fuel in the reactor core will be used, the range of the hazards for this experimental program can be determined by studying the kinetics of a few selected core loadings representative of the extremes of enrichment and critical moderator levels.

b. Physics Constants for Various Fuel Loadings

The inherent safety of a core is determined by those physics characteristics which reduce the reactivity of the core during an excursion transient. Those physics constants which are most important to inherent safety are the Doppler coefficient, moderator void coefficient, and moderator temperature coefficient. Since the delayed neutron fraction and the neutron lifetime are important in determining the severity of an excursion arising from a given input of reactivity, they, too must be considered in analysis of the nuclear hazards. Calculated values for the Doppler coefficient, moderator void coefficient, moderator temperature coefficient, delayed fraction, and neutron lifetime are given in Table 1. 1 for various PRCF fuel loadings.

TABLE 1. 1

PHYSICS CONSTANTS FOR VARIOUS FUEL LOADINGS

Core Loading			Moderator Void Coefficient, $\Delta k/k/\%$ Void		Moderator Temperature Coefficient, $\Delta k/k/F$, 70 to 210 F		
Fuel Arrangement	Critical Moderator Level, ft	Coefficient, $\Delta k/k/F$*	Coolant	Moderator		β	ℓ*
All Pu-Al	5	0	1.07×10^{-4}	4.60×10^{-4}	1.45×10^{-5}	2.51×10^{-3}	$0.35 - 0.40 \times 10^{-3}$
All Pu-Al	9	0	1.45×10^{-4}	7.65×10^{-4}	2.32×10^{-5}	2.51×10^{-3}	0.35×10^{-3}
Zoned	5	0.23×10^{-5}	1.07×10^{-4}	4.60×10^{-4}	1.45×10^{-5}	3.39×10^{-3}	0.40×10^{-3}
Zoned	9	0.51×10^{-5}	1.15×10^{-4}	4.60×10^{-4}	1.47×10^{-5}	4.27×10^{-3}	0.45×10^{-3}
Quasi-uniform	9	0.84×10^{-5}	2.2×10^{-4}	10.35×10^{-4}	3.20×10^{-5}	5.34×10^{-3}	0.45×10^{-3}

*For temperatures from 100 to 500 F.

The range of critical moderator levels permitted in the PRCF will be 5 to 9 ft. Values for core loadings calculated to give critical moderator levels of either 5 or 9 ft are given in Table 1. 1, representing the extremes. Of the D_2O moderated cores planned for the PRCF, a core containing only Pu-Al fuel elements will have the smallest delayed neutron fraction and essentially no negative fuel temperature coefficient; and the quasi-uniform core with a 9 ft critical moderator level will have the largest delayed neutron fraction and the strongest negative fuel temperature coefficient.

Physics constants for cores other than those listed in Table 1.1 should fall between the extremes of the values given in the table, or additional analyses will be made.

c. Analysis of Control Errors

(1) General

Control errors leading to continued withdrawal of control or safety rods or continued addition of moderator with the reactor critical could lead to nuclear excursions. This type of accident could occur either when the reactor is being started up or when it is operating at constant power level. Such accidents were analyzed for five different core loadings, representing the extremes of critical moderator levels and levels of enrichment. The core loadings studied were:

- All Pu-Al Fuel Loadings

5 ft critical moderator level; 55 fuel elements
9 ft critical moderator level; 13 fuel elements

- Zoned Fuel Loadings

Zoned fuel loadings consist of discrete regions of Pu-Al fuel elements and UO_2 fuel elements.

5 ft critical moderator level; central region of 9 UO_2 fuel elements surrounded by 46 Pu-Al fuel elements. It was estimated that 80% of the fissions would occur in the plutonium.

9 ft critical moderator level; central region of 21 UO_2 fuel elements surrounded by 34 Pu-Al fuel elements. It was estimated that 60% of the fissions would occur in the plutonium.

- Quasi-uniform Fuel Loading

A quasi-uniform core loading is one in which the UO_2 and the Pu-Al fuel elements are distributed as uniformly as possible throughout the core. A quasi-uniform fuel loading of 19 Pu-Al fuel elements and 36 UO_2 fuel elements would give a critical moderator level of 9 ft. It was estimated that 35% of the fissions would occur in the plutonium.

Excursions resulting from control errors were simulated on an analog computer. The equations used in the mathematical model of the reactor are given in Appendix B. Basic assumptions used in these studies were:

- The excursions were initiated by a positive ramp input of reactivity of 10¢/sec which is the operating limit for control devices.
- Failure of period trips was assumed for all cases.
- The excursions were terminated either by an automatic safety circuit trip when the power level increased to 150 W or by the inherent shut-down mechanisms and manual breaking of the rod magnet circuit when it was assumed that the safety circuit failed to trip.

(2) Excursions Terminated by Safety Circuit Trip

(a) Startup Accidents

The startup accidents were initiated with the reactor approximately 0.5 mk subcritical by adding reactivity at a rate of 10 ¢/sec until 1 $ of positive reactivity had been added. This type of startup accident was analyzed for each of the five fuel loadings described above. When the safety circuit was tripped at a power level of 150 W, it was assumed that only one of the safety rods was dropped; the worth of one rod was 25 mk. The insertion of negative reactivity by dropping one safety rod is shown in Figure 1.1 for both 9- and 5-ft critical moderator levels.* The power level transients and the integrated energy release for startup accidents for the five different core loadings are shown in Figure 1.2. In no case did the power increase to a level that would endanger the reactor nor would the calculated energy release significantly increase the temperature of the core. The average temperature of the fuel did not increase perceptibly in any of the cases.

A startup accident in which the addition of reactivity at 10 ¢/sec was continued until the safety circuit was tripped was also studied for a

*The negative reactivity insertion rates shown in Figure 1.1 differ from those shown in Figure 19, HW-69168, because the rod strength used in this report is 25 mk instead of 35 mk. A rod strength of 25 mk agrees with the minimum safety rod strength proposed as an operating limit.

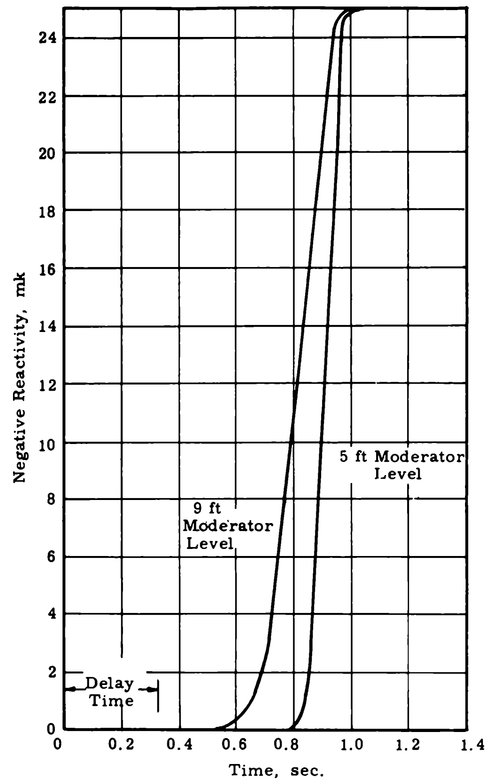
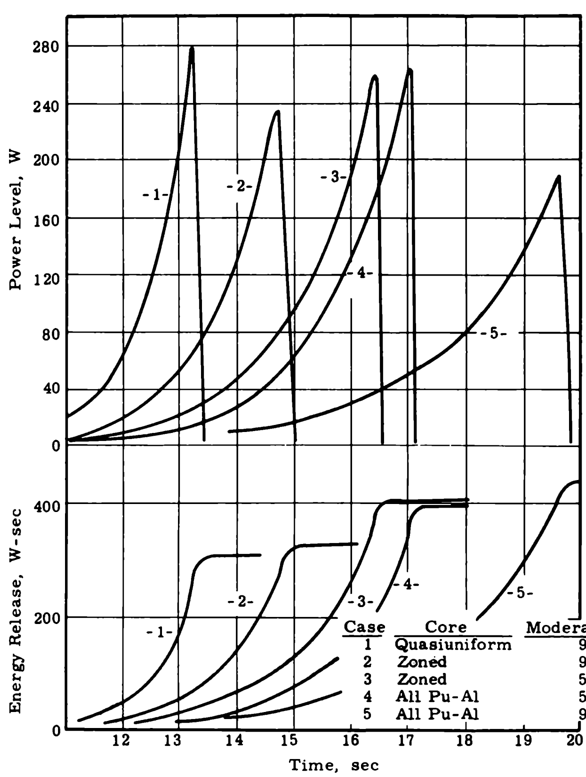


FIGURE 1.1

Scram Curves for Reactor with Moderator at 9- and 5-ft Levels



Case	Core	Moderator Level, ft	mk/sec
1	Quasiuniform	9	0.534
2	Zoned	9	0.427
3	Zoned	5	0.339
4	All Pu-Al	5	0.251
5	All Pu-Al	9	0.251

FIGURE 1.2

Startup Accident for Various Core Loadings zoned core with 5 ft critical moderator level. The power level transient and the energy release are compared in Figure 1. 3 with the results of a startup accident in a similar core with reactivity addition at 10 ¢/sec ended when 1 $ of positive reactivity had been added. The continued addition of reactivity resulted in a higher peak power and total energy release, but even in this case the reactor was shut down well before any damage could occur.

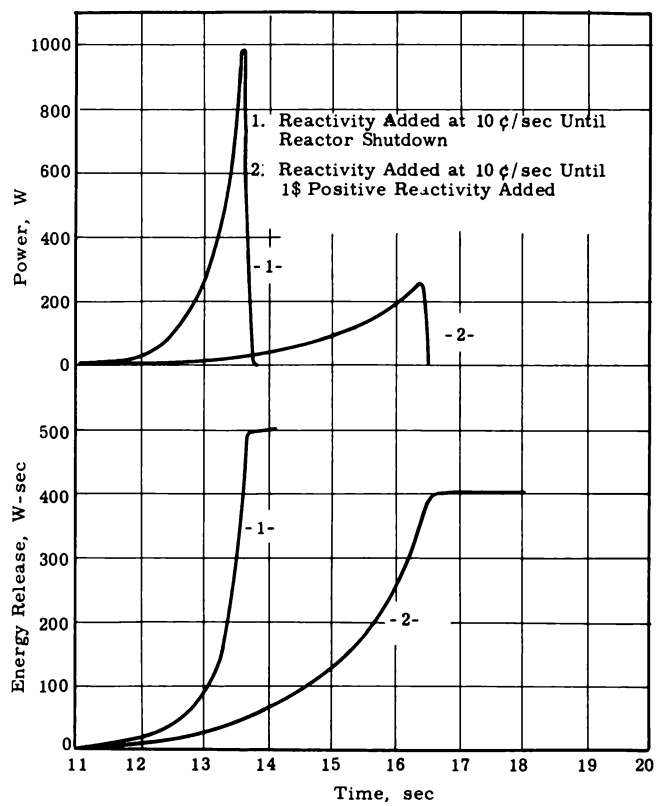


FIGURE 1. 3

Startup Accident: Effect of Continued Reactivity Addition

(b) Control Error at 100 W

Also studied were accidents that could result from the control error of adding reactivity at 10 ¢/sec until 1 $ of positive reactivity had been added while the reactor was operating at a power level of 100 W. The safety circuit was tripped at a power level of 150 W and negative reactivity was inserted by one rod dropping as indicated in Figure 1. 1. The results of these studies for the five different core loadings are shown in Figure 1. 4. Power levels attained were not high enough to cause core damage, and the energy release in none of the accidents was significantly greater than the energy that would have been generated by continued operation at 100 W.

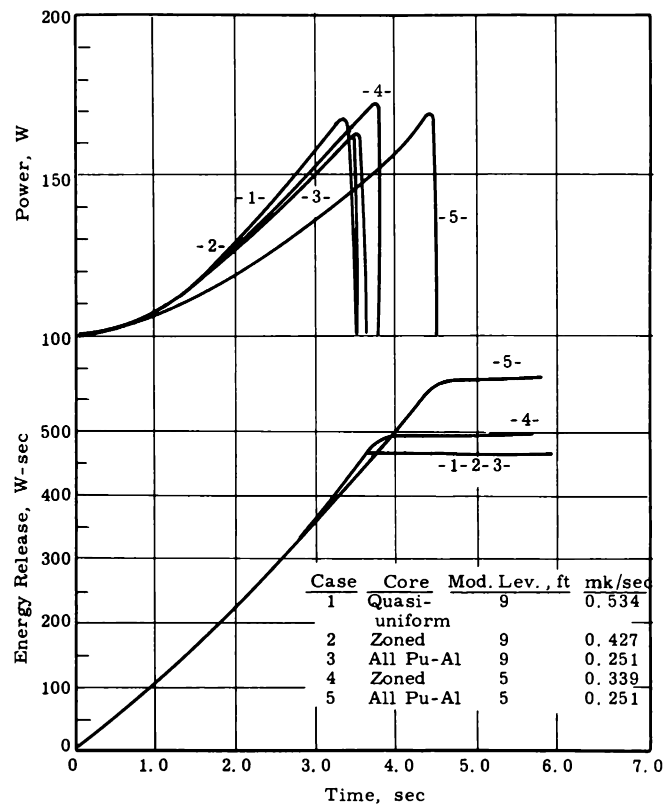


FIGURE 1. 4

Control Error for Various Core Loadings from 100 W

(3) Excursions Not Terminated by Safety Circuit Trip

(a) Accident Analysis

In the event that the safety circuit should fail to trip, the inherent shutdown mechanisms of the reactor core will eventually override an excursion. A series of analog computer studies of simulated excursions assuming that the safety circuit failed to trip, with core loadings identical to those described above, confirm this assertion.

The principal inherent shutdown mechanisms of the PRCF cores are:

- Enhanced resonance absorption due to Doppler broadening of the U^{238} neutron absorption peaks as the fuel temperature is increased,

- Void formation in the moderator by heat transfer and radiolytic decomposition of the moderator.

The negative fuel temperature coefficient is discussed in HW-69168, p. 56. In the reactor kinetics studies described here, an overall core negative fuel temperature coefficient was estimated for each core loading (see Table 1. 1). In the analog studies, the core average UO_2 fuel temperature was calculated, converted to negative reactivity, and fed back to the reactivity circuit.

Since the PRCF core is a D_2O moderated assembly of PRTR fuel elements arranged in a lattice spacing identical to PRTR, the PRCF core was analyzed in the same manner as the PRTR core, considering the moderator as though it were present in two regions, the "coolant" and the "moderator," even though there are no discrete coolant channels in the PRCF. The "coolant" region was defined as that D_2O immediately surrounding each fuel element and the "moderator" region defined as the remainder of the D_2O in the reactor core. The void coefficients used in these analog studies were calculated using a computer code(1) developed for evaluation of the void effect upon loss of coolant from a PRTR process tube. "Coolant" and "moderator" void coefficients for the five core loadings studied are given in Table 1. 1.

Void formation by heat transfer is a function of the fuel surface temperature; and it was assumed, for Pu-Al fuel elements, that the surface temperature would be the same as the fuel element core temperature because of the high thermal conductivity of aluminum. The void formed by heat transfer from the fuel elements would occur as the result of film and nucleate boiling on the fuel element surface (the maximum thickness of the calculated steam void around the fuel rods was 0.010 in. for any of the cases studied).

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(1) Adapted from Program F-3 - IBM 704, Three-Group Neutron Diffusion Calculation, J. G. Keppler, et al., by J. R. Lilley, Jr. and J. J. Regimbal.

Equations used to determine the volume of vapor formed by heat transfer were formulated from a steam void model for vertical plane surfaces described by Janssen, et al.[1]

Assumptions made for the calculation of the heat transfer void were:

- The steam film would be formed immediately after the onset of boiling.
- The heat required to raise the coolant film to the saturation temperature and the heat of vaporization of the coolant film were neglected because the heat involved would be an insignificant fraction of the heat transferred across the steam film.
- The volume of the steam film surrounding the fuel rods would be proportional to the fuel core temperature and was calculated as shown in Appendix B.
- The time required to override the excursion would be so short that the rise in the bulk moderator temperature would be negligible.
- The thermal properties of D_2O would be constant throughout the transient.

The power level transient, integrated energy release, and maximum Pu-Al fuel element temperature are shown in Figure 1. 5 for the five different core loadings for an accident occurring when the reactor is operating at 100 W. These accidents were initiated by the addition of 1 $ of excess reactivity at a rate of 10 ¢/sec. Power levels reached in these accidents were much greater than those in the accidents terminated by a safety circuit trip. Only in the quasi-uniform and all Pu-Al cores with 9 ft critical moderator level did the maximum Pu-Al fuel element temperature reach or exceed the melting point of aluminum.

In none of these accidents was the combined effect of the negative fuel temperature coefficient and the coolant void coefficient great enough to shut down the reactor after the excursion had been arrested. After the peak of the

(1) E. Janssen, W. H. Cook and K. Hikido. Metal-Water Reactions: I. A Method for Analyzing a Nuclear Excursion in a Water Cooled and Moderated Reactor, GEAP-3073. October 15, 1958.

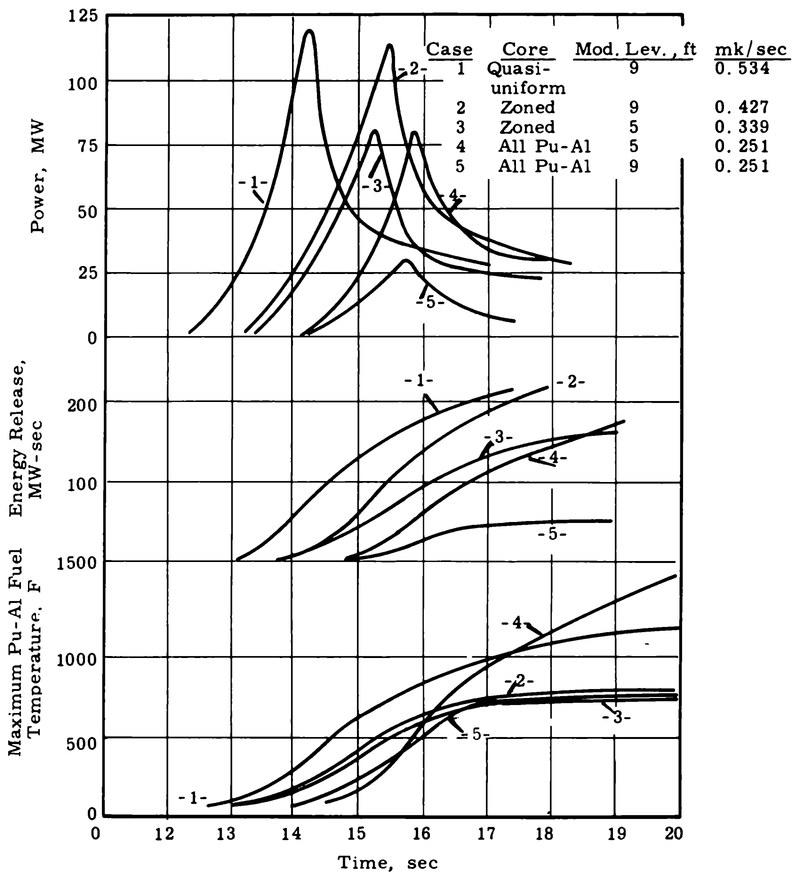

FIGURE 1. 5

Control Error at 100 W: Safety Circuit Fails to Trip

excursion had been passed, the power level was determined by the magnitude of the negative temperature and void coefficients characteristic of the particular core loading. A means of adding negative reactivity is needed in addition to that provided by the negative fuel temperature and void coefficients. Breaking the rod magnet circuit by a button on the control console will drop the safety and control rods and shut down the reactor.

A reactivity addition rate of 10 ¢/sec results in a relatively slow excursion. The safety circuit trip point, 150 W, is not reached until about 3 sec after the start of reactivity addition to the reactor when operating at 100 W. Furthermore, the peak of the excursion is not attained until 14 to 16 sec has elapsed. A well trained console operator should react to a situation such as this rapidly enough to deenergize the rod magnet circuit before the power level peak has been reached, thus limiting the severity of the accidents described above.

A startup accident and an accident occurring while operating at 100 W power level are compared in Figure 1. 6. In both cases, reactivity was added at 10 ¢/sec until 1 $ excess reactivity had been added. The core loading studied for this comparison was the all Pu-Al core with 9 ft critical moderator level. This comparison shows little difference in the severity of the two accidents.

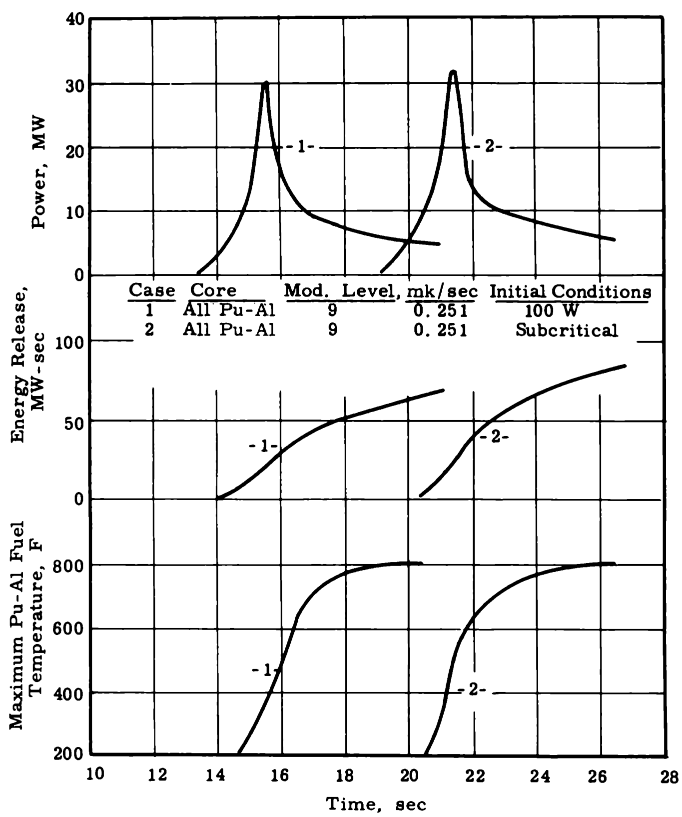

FIGURE 1. 6

Comparison of Startup Accident with Accident of 100 W Power Level: Safety Circuit Fails to Trip

The effect on the severity of an accident of varying amounts of excess reactivity slightly greater than 1 $ was also investigated. The all Pu-Al core with 9 ft critical moderator level was considered in this analysis. Simulated accidents were initiated at a power level of 100 W by adding reactivity at 10 ¢/sec until the total excess reactivity added in the different cases was 1. 0, 1. 1, 1. 2, and 1. 5 \$. In these accidents it was assumed that the safety circuit failed to trip. The power level transient, energy release, and maximum Pu-Al fuel temperature are shown in Figure 1. 7. The results indicate that the negative reactivity effect of void formation is not strong enough to prevent melting of the fuel if the excess reactivity added is slightly greater than 1 \$. However, prompt action by the console operator in deenergizing the rod magnet circuit could terminate the excursion before fuel temperatures increased to the melting point.

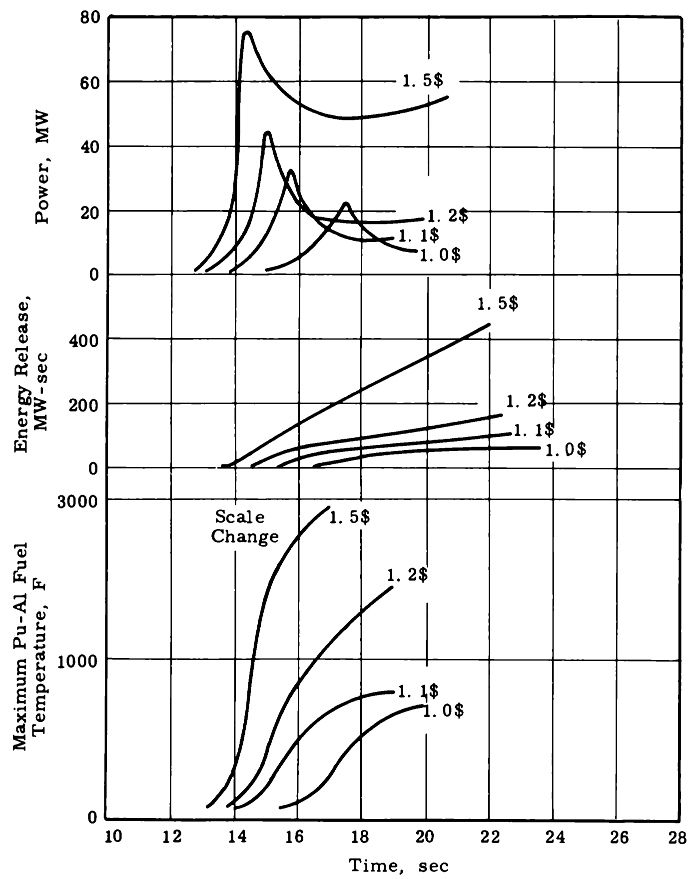

FIGURE 1. 7

Effect of Excess Reactivity on Severity of Excursion: Safety Circuit Fails to Trip

(b) Radiological Consequences

None of the accidents described above could result in a release of fission products significantly greater than the release in the maximum credible accident described in HW-69168, because none of the excursions generate a fission product inventory in the core which is significant when compared to the fission product inventory of a preirradiated fuel element. Also, the maximum energy release in these accidents, about 400 MW sec in about 20 sec, would raise the moderator temperature only about 20 F. However, the radiological consequences of the maximum credible accident have been further analyzed, with particular consideration being given to means of minimizing or eliminating leakage of radionuclides from the reactor cell and the hazards posed by the presence of plutonium isotopes in the cell leakage and the discharge from the PRTR stack.

Leakage rate tests of the reactor cell performed after installation of all of the equipment demonstrated that the lowest leakage rate that could be consistently maintained is 1200 ft^3/day with the cell pressure at 6 in. of water. This rate was achievable only after concerted effort to locate and seal leaks. With the cell pressurized to 6 in. of water and with helium injected into the cell, no leaks were identified with a helium detector that could account for the 1200 ft^3/day leakage rate. This rate is significantly greater than the design criteria maximum leakage rate of 400 ft^3/day at a cell pressure of 2 psig.

In operating a confinement system, it is highly desirable to have no unfiltered leakage from the confinement cell so that radionuclide contamination of the environs can be held to a minimum in the event of an accident. The most feasible method of preventing unfiltered leakage from the PRCF cell is to mechanically maintain a slightly negative cell pressure, with reference to atmospheric pressure, by continuous operation of the exhaust fan. The exhaust air piping arrangement is shown on the simplified piping diagram, Figure 1. 8. As initially designed, the PRTR containment-PRCF confinement system automatically stopped the PRTR exhaust fan upon a containment trip. This action of the automatic containment circuit will be eliminated to provide continuous operation of the exhaust fan. Emergency power backup from the PRTR diesel generator will be supplied to the fan during failure of the normal power supply. With this system there will be a very low probability that the reactor cell pressure could be positive during a reactor accident.

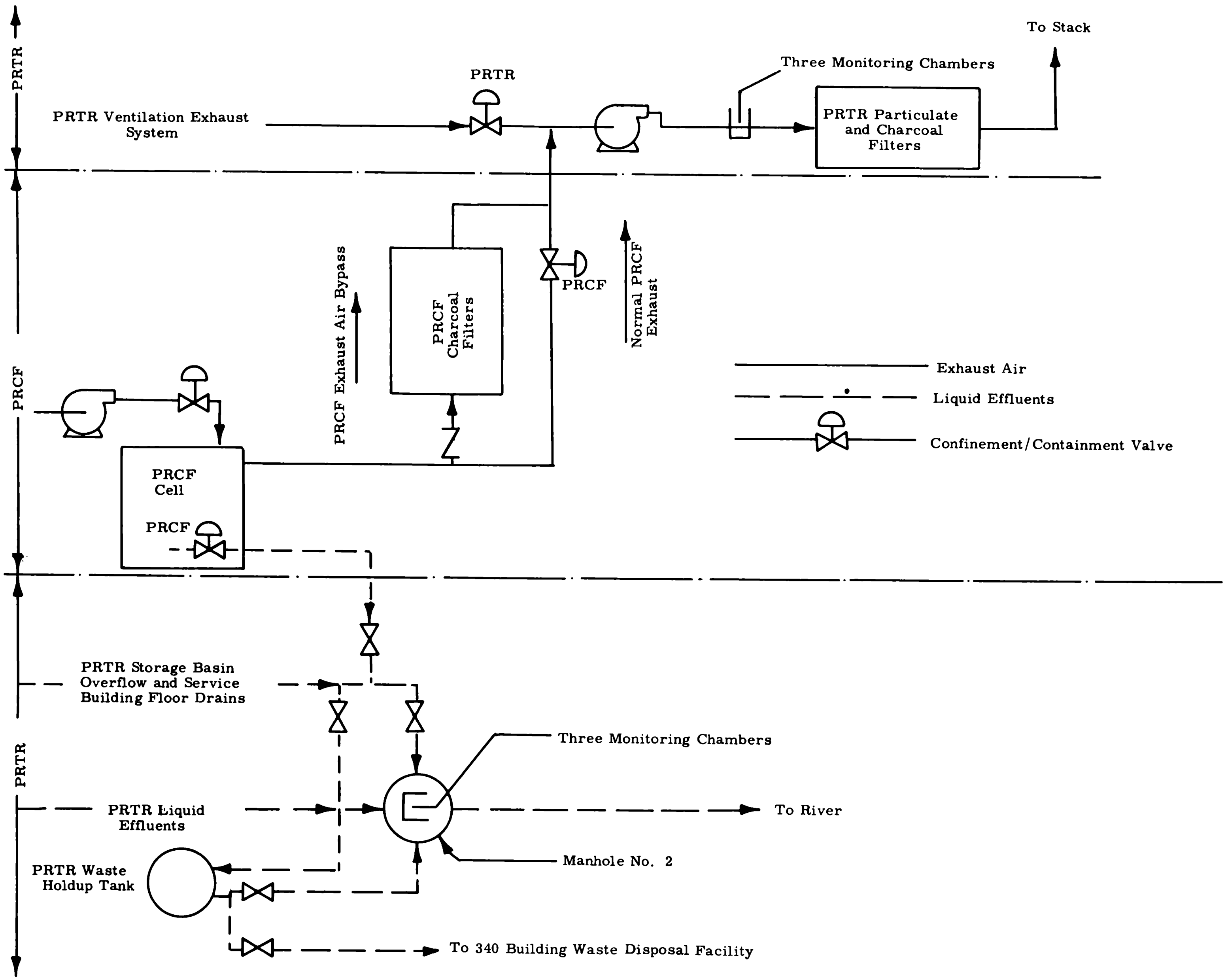

FIGURE 1. 8

PRTR-PRCF Effluent Activity System Simplified Piping Diagram

For the maximum credible accident, melt-down of a short-cooled Pu-Al fuel element, described in HW-69168, p. 88, the calculated maximum rate of vapor flow from the cell required to maintain a constant pressure in the cell is 50 ft^3/min. The PRTR exhaust fan has ample capacity to maintain a negative pressure of about 3 in. of water in the cell during confinement.

The quantities of radionuclides in the Pu-Al fuel element and the quantities released to the reactor cell are given in Tables IV and V of HW-69168. In calculating the escape of radionuclides from the confinement cell by either the stack route or unfiltered leakage, the following plate-out factors (fraction of the radionuclides escaping from the cell) were used.

Noble Gases	1. 0
Halogens	0. 5
Volatile Solids	0. 3
Nonvolatile Solids	0. 3

With the PRTR exhaust fan operating continuously, a slightly negative pressure is maintained in the PRCF cell, even during the course of the maximum credible accident. During PRCF confinement, all vapor leaving the PRCF cell is routed through the PRCF charcoal filters, the PRTR particulate and charcoal filters, and out of the 150 ft high stack. The PRCF activated charcoal filters, two in series, will give a halogen removal efficiency of 99. 5% and the PRTR particulate filter efficiency is 99. 95% for removal of all particles 0. 3 μ and larger. The quantities of radionuclides that would be released from the stack are given in Table 1. 2.

TABLE 1. 2

QUANTITIES OF RADIONUCLIDES RELEASED FROM THE STACK, Ci

Radionuclides	Decay Time 3 hr*	Decay Time 24 hr
Noble Gases	1.8×10^5	1.0×10^5
Halogens	250	170
Volatile Solids	13	8. 5
Nonvolatile Solids	2. 8	1. 5
Total Fission Products	1.8×10^5	1.0×10^5
Pu^{239}	1.3×10^{-7}	1.3×10^{-7}
Pu^{240}	1.7×10^{-7}	1.7×10^{-7}
Pu^{241}	1.7×10^{-5}	1.7×10^{-5}
Pu^{242}	-----	-----

*3 hr is considered to be the shortest time after PRTR shutdown in in which a fuel element could be transferred from PRTR to PRCF.

An evaluation of the whole-body and thyroid doses for an adult remaining at the centerline of the cloud as it passed was made for several atmospheric conditions. Estimates were made for strong inversion, moderate inversion, and neutral temperature gradients with a ground wind speed of 1 m/sec. In addition, the maximum lung uptake of plutonium (practically all Pu^{241}) which would be received was calculated to be much less than 10^{-5} μCi at all distances greater than 500 m from the facility. The whole-body and thyroid doses are given in Table 1.3 for various distances from the facility.

TABLE 1.3

ESTIMATED RADIATION DOSES ON CENTERLINE OF CLOUD DOWNWIND

(All vapor released through stack)

	Atmospheric Stability[(a)]					
	Strong Inversion		Moderate Inversion		Neutral	
Distance, m	Whole[(b)] Body, r	Thyroid,[(c)] rad	Whole[(b)] Body, r	Thyroid,[(c)] rad	Whole[(b)] Body, r	Thyroid,[(c)] rad
500	$< 10^{-3}$	$< 10^{-3}$	$< 10^{-3}$	$< 10^{-3}$	2.4	1.5
1,000	$< 10^{-3}$	$< 10^{-3}$	0.02	$< 10^{-3}$	2.9	1.8
5,000	$< 10^{-3}$	$< 10^{-3}$	0.2	$< 10^{-3}$	0.3	0.2
10,000	$< 10^{-3}$	$< 10^{-3}$	0.2	0.1	0.1	0.06
50,000	2×10^{-3}	$< 10^{-3}$	0.05	0.02	0.006	0.003
100,000	7×10^{-3}	$< 10^{-3}$	0.03	0.005	0.002	$< 10^{-3}$

(a) Ground wind speed assumed to be 1 m/sec.
(b) The body dose accumulated during cloud passage.
(c) The thyroid dose from inhalation during cloud passage.

No significant biological effects would be expected as a result of the doses shown in Table 1.3. Estimates of the areas which would be contaminated to the level of 1 μCi I^{131}/m^2 are given in Table 1.4. Confiscation of milk and leafy vegetable produce might be required in these small areas.

TABLE 1.4

AREAS CONTAMINATED TO 1 μCi I^{131}/m^2

(All vapor released through stack)
(Ground wind speed 1 m/sec)

Atmospheric Stability	Area, mi^2
Strong Inversion	0
Moderate Inversion	0.3
Neutral	0.2

The maximum credible accident in the PRCF could result in a positive pressure in the reactor cell only if the PRTR exhaust fan should fail simultaneously. This would require the compounding of one or two additional simultaneous failures in excess of those leading to the maximum credible accident. These additional failures are mechanical failure of the exhaust blower or failure of power followed by failure of the PRTR emergency diesel generator. If the blower should fail, there would be some unfiltered leakage from the cell at ground level as well as the filtered release of vapor from the stack. In estimating the consequences for this case, it was assumed that during the maximum credible accident 50 ft^3/min of vapor was discharged through the exhaust system and that unfiltered leakage from the cell occurred at a rate of 1200 ft^3/day. With these assumptions, 50 ft^3 of unfiltered gas would be released at ground level; and the remainder of the radionuclides, that did not plate out, would be transported from the cell by vapor flowing through the filters and discharged from the stack. The quantities of radionuclides that would be released from the stack and at ground levels are given in Table 1. 5.

TABLE 1. 5

QUANTITIES OF RADIONUCLIDES RELEASED FROM REACTOR CELL, Ci

	Decay Time			
	Stack Release		Ground Release	
Radionuclides	3 hr[a]	24 hr	3 hr[a]	24 hr
Noble Gases	1.8×10^5	1.0×10^5	2.6×10^3	1.4×10^3
Halogens	250	170	7.2×10^2	4.9×10^2
Volatile Solids	13	8.5	2.2×10^2	1.4×10^2
Nonvolatile Solids	2.8	1.5	47	29
Total Fission Products	1.8×10^5	1.0×10^5	3.6×10^3	2.0×10^3
Pu^{239}	1.3×10^{-7}	1.3×10^{-7}	2.6×10^{-4}	2.6×10^{-4}
Pu^{240}	1.7×10^{-7}	1.7×10^{-7}	3.4×10^{-4}	3.4×10^{-4}
Pu^{241}	1.7×10^{-5}	1.7×10^{-5}	3.4×10^{-2}	3.4×10^{-2}
Pu^{242}	-----	-----	-----	-----

(a) Three hours is considered to be the shortest time after PRTR shutdown in which a fuel element could be transferred from the PRTR to the PRCF.

In Table 1. 6 the estimated doses on the centerline of the cloud downwind of the facility are given for several atmospheric conditions. The maximum lung uptake of plutonium (practically all Pu^{241}) was calculated to be less than 5×10^{-3} μCi at all distances greater than 500 m.

TABLE 1.6

ESTIMATED RADIATION DOSES ON CENTERLINE OF THE CLOUD DOWNWIND

(Release from stack and at ground level)[a]

	Atmospheric Stability[b]					
	Strong Inversion		Moderate Inversion		Neutral	
Distance, m	Whole Body,[c] r	Thyroid,[d] rad	Whole Body,[c] r	Thyroid,[d] rad	Whole Body,[c] r	Thyroid,[d] r
500	2	140	0.8	70	3	14
1,000	1	60	0.4	25	3	5
5,000	0.2	5	0.2	3	0.4	0.3
10,000	0.06	1	0.2	1	0.1	0.1
50,000	0.01	$<10^{-3}$	0.05	0.03	0.006	0.03
100,000	0.01	----	0.03	0.005	0.003	$<10^{-3}$

(a) 50 ft^3/m flowing through exhaust system filters to stack. 50 ft^3 unfiltered vapor released at ground level.
(b) Ground wind speed assumed to be 1 m/sec.
(c) The body dose accumulated during cloud passage.
(d) The thyroid dose from inhalation during cloud passage.

From the calculated doses shown in Table 1.6, evacuation would not be necessary for any off-site areas. However, deposition of radioiodine could contaminate to a level of 1 μCi/m^2 the areas shown in Table 1.7 for different weather conditions. This might require action to confiscate milk and leafy vegetable crops in relatively small areas.

TABLE 1.7

AREA CONTAMINATED TO 1 μCi I^{131}/m^2

(Release from stack and at ground level)
(Ground wind speed 1 m/sec)[a]

Atmospheric Stability	Area, mi^2
Strong Inversion	5
Moderate Inversion	5
Neutral	0.5

(a) 50 ft^3/min flowing through exhaust system filters to stack. 50 ft^3 unfiltered vapor released at ground level.

2. Air Cooling of Irradiated Pu-Al Fuel Element

The temperature transient for an irradiated Pu-Al fuel element cooled by free convection in air is shown in Figure 23, HW-69168. It is estimated that the maximum core temperature of the central fuel rod during this transient would be 1655 F (~900 C). It is improbable that any of the other rods of the fuel element would melt. Although this maximum temperature is well below the melting point of the Zircaloy-2 jacket, it is possible that expansion of the molten core and the fission gas pressure would rupture the jacket.

The mechanical properties of Zircaloy-2 decrease markedly at temperatures above 700 to 800 F. For example, the slope of the curve of ultimate tensile strength versus temperature is negative and increasing as the temperature is increased above 800 F[1] as indicated by data from Knolls Atomic Power Laboratory. Although there are no data given in the reference for temperatures above 1100 F, the ultimate tensile strength is decreasing so rapidly with increasing temperature that extrapolation of the curve indicates that jacket failure of a Pu-Al fuel element rod should occur when the Pu-Al core has melted (1220 F).

Release of the molten core from the central rod of an irradiated Pu-Al fuel element hanging in air would lead to a release to the cell of radionuclides significantly less than the maximum credible accident described in HW-69168. Furthermore, this accident would generate less energy, since the probability of metal-water reaction would be much lower, and less metal would react even if the reaction did occur. Since the discharge rate through the confinement system filters, the PRCF cell pressure, and the quantity of radionuclides released from the fuel would all be lower, the radiological consequences would be much less severe than described for the maximum credible accident.

3. Flexible Hoses for Irradiated Fuel Element Transfer Thimble

The flexible hoses used for circulating coolant through the short-cooled fuel element transfer thimble are fabricated of convoluted 300 series stainless steel, with a stainless steel braid covering to restrict the amount of flexing of the convolutions. These hoses are rated for 730 psig service at room temperature. The vendor certified the PRCF hoses for service at 150 psig and 250 F and stated in the certification that the hoses were shop tested under conditions in excess of those certified.

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(1) Selected Mechanical Properties of Cladding and Structural Materials, Reactor Core Materials, vol. 2, no. 1, p. 34. February, 1959.

A maximum pressure of 45 psig can be developed in the thimble coolant loop by the shut-off head of the circulating pump, although the operational error of closing the inlet and outlet valves (V-1 andV-4 in Figure 6, HW-69168) could result in a pressure increase to 110 psig, the bursting pressure of the rupture disc. The section of the loop between the inlet and outlet valves has been pressure tested at 125 psig at room temperature. This pressure test will be repeated before the first short-cooled irradiated fuel element is transferred into the cell.

4. PRTR-PRCF Safety and Containment Circuit Interties

a. Effluent Systems

The aqueous effluents and exhaust ventilation air from the PRCF are discharged to the respective PRTR effluent system. As shown in Figure 1. 8, exhaust air from the PRCF is routed to the PRTR ventilation system and is monitored for activity by the PRTR monitoring chambers. Aqueous wastes from the PRCF are discharged to the PRTR aqueous waste disposal system. As shown in Figure 1. 8, PRTR storage basin overflow is routed through the the same line as PRCF aqueous wastes and PRCF wastes can be diverted to the PRTR waste hold-up system. Aqueous PRCF wastes which are routed to the river are monitored by the PRTR monitoring chambers installed in Manhole No. 2.

These aqueous and exhaust air monitoring chambers are also the sensing devices to provide the primary signals for PRTR containment and PRCF confinement. Since the PRCF and the PRTR share the same activity monitors, activity released from either facility in sufficient quantities to reach the trip (containment) point will initiate automatic action in both facilities. Automatic action upon a containment/confinement trip is summarized in Table 1. 8. The relative locations of effluent monitor indicators and recorders and annunciators associated with the effluent monitor and containment/confinement systems are shown in Figure 1. 9. Annunciators are located at the control console of each of the facilities.

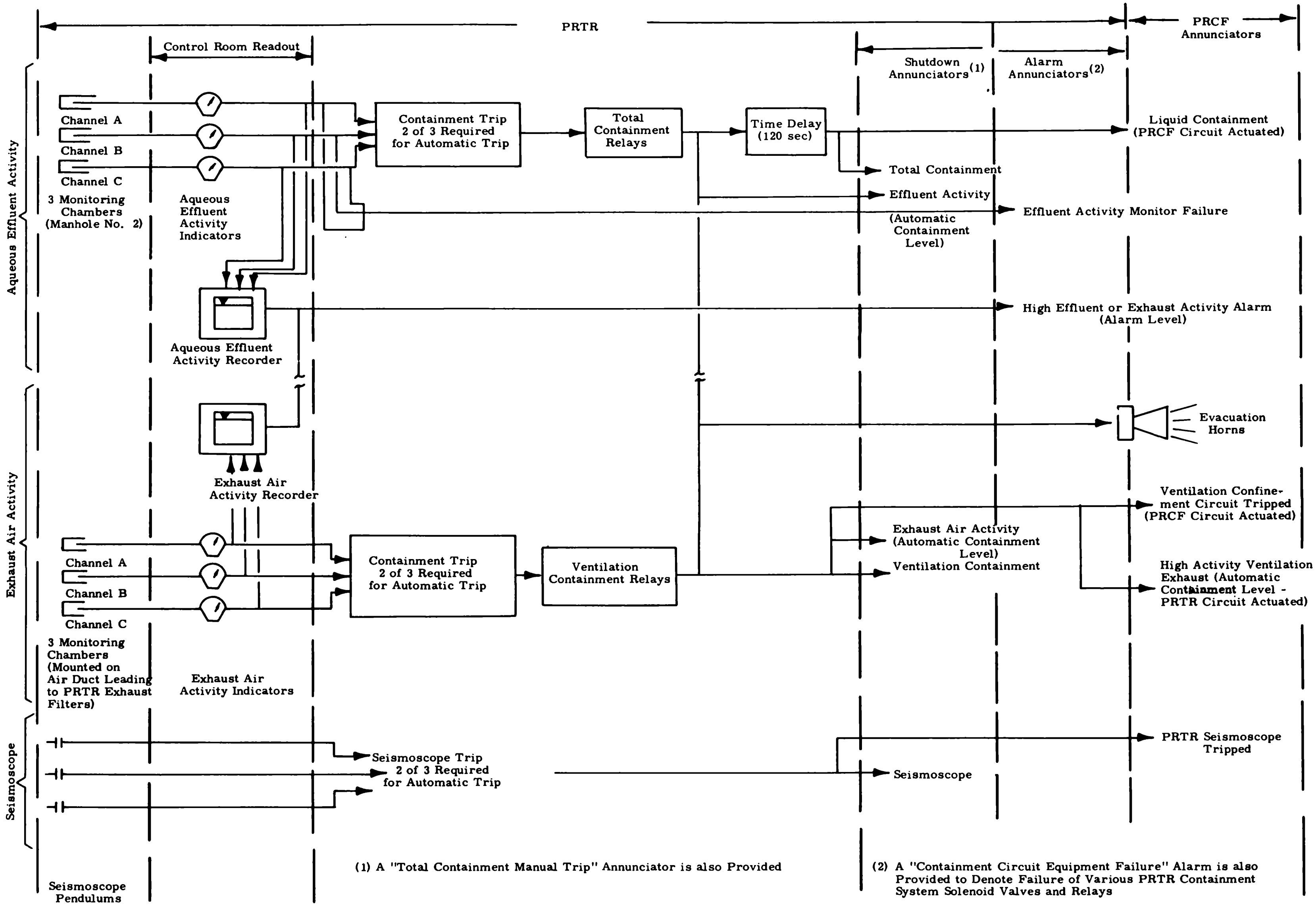

FIGURE 1.9

PRTR - PRCF Effluent Activity and Seismoscope Alarm and Monitor Interties

TABLE 1. 8

AUTOMATIC ACTION OF PRTR-PRCF COMPONENTS UPON A CONTAINMENT/CONFINEMENT TRIP

Trip Signal	PRTR	PRCF
2 of 3 aqueous effluent high activity trips activated	1. Reactor scrams 2. Aqueous and ventilation containment valves close after time delay of 120 sec.	1. Reactor scrams 2. Cell supply fan stops 3. Aqueous and ventilation confinement valves[a] close after time delay of 120 sec.
2 of 3 exhaust air high activity trips activated	1. Reactor scrams 2. Ventilation containment valves close	1. Reactor scrams 2. Cell supply fan stops 3. Ventilation confinement valves close.[a]

(a) A bypass line around the PRCF exhaust air confinement valve in the effluent line (Figure 1. 8) is provided to prevent pressure buildup in the PRCF reactor cell. The bypass line is equipped with activated charcoal filters. The filtered air is then routed to the PRTR stack via the PRTR particulate and charcoal filters.

b. Other Interties

The PRCF safety circuit is activated by a PRTR seismoscope trip (the PRTR seismoscope consists of one high sensitivity and two low sensitivity pendulums; actuation of any two of the three channels is required for safety circuit trip). An annunciator, shown in Figure 1. 9, is provided at the PRCF console to indicate that a seismoscope trip has been received.

The radiation levels in the PRCF reactor cell and operating area are recorded in the PRTR control room as well as indicated at the PRCF console.

The PRCF safety circuit is energized by the PRTR 125 V dc system. System variables are indicated in the PRTR control room; e. g., resistance to ground and voltage. It should be noted that the dc supply for the rod magnet circuit is independent of the PRTR 125 V system.

The PRCF safety circuit will be locked out whenever the facility is left unattended. Any off-standard condition that is annunciated in the PRCF after the safety circuit is locked out will actuate a single annunciator (PRCF Off-Standard) at the PRTR console.

5. Emergency Planning

a. Minimizing Operational Mishaps

In HW-69168, it was indicated that operational mishaps will be minimized by design safety features, such as the safety system and interlocks, and by effective administrative controls, such as process specifications and operating procedures. A more detailed discussion of the administrative provisions for minimizing operational mishaps is given below.

Administrative provisions which have been established are process specifications, operating procedures, maintenance procedures, test descriptions, test procedures, personnel training and qualification, and control of design changes.

Process Specifications

- Prepared by Nuclear Safety Studies Operation (NSSO)
- Accepted by Manager, Test Reactor and Auxiliaries Operation (TRAO)
- Approved by Manager, Reactor and Fuels Laboratory (RAFL)
- Operational compliance audited by NSSO, RAFL, and Technical Planning Operation (TPO), TRAO

Operating Procedures

- Prepared by Plutonium Recycle Critical Facility Operation (PRCFO), TRAO
- Approved by Supervisor, PRCFO
- Audited by PRCFO

Maintenance Procedures

- Prepared by Maintenance and Equipment Engineering Operation (MEEO), TRAO
- Accepted by PRTR Maintenance Operation (PRTRMO), TRAO
- Approved by Supervisor, PRCFO
- Audited by MEEO and PRCFO

Test Descriptions

- Prepared by test sponsor
- Approved by Supervisor, PRCFO; Supervisor, TPO; and Manager, TRAO
- Reviewed by NSSO

Test Procedures

- Prepared by PRCFO
- Reviewed by test sponsor
- Approved by Supervisor, PRCFO
- Audited by TPO

Personnel Training and Qualification

- Conducted by PRCFO
- Oral and demonstration tests administered by PRCFO
- Written tests administered by TPO

Design Changes

Process design changes are controlled and documented as follows:

- Prepared by MEEO
- Approved by Supervisor, MEEO; Supervisor, TPO; Supervisor, PRCFO; Manager, NSSO; and Manager, TRAO; as well as other interested groups, such as Radiation Protection and PRTR Operation, where appropriate.

b. Investigating Unusual or Unexpected Incidents

Investigation and reporting of incidents in the PRCF will conform to existing HAPO practice. This will include compliance with directives requiring prompt reporting of unusal events. Formal investigations will be conducted by ad hoc committees constituted of expert and disinterested personnel at an organizational level appropriate to the seriousness of the incident. All operational experience including investigations of minor incidents (troubleshooting) will be recorded in the operating log books.

c. Accident Recovery Techniques

In the event of a maximum credible accident as described in HW-69168, it was indicated that off-site contamination would be limited to relatively insignificant levels. Serious contamination would be restricted to the PRCF cell and the exhaust air line. Adequate shielding is provided around the reactor cell to permit extended access to the PRCF operating area.

Within the controlled access area surrounding the PRCF site (minimum distance from facility, 4000 ft), a maximum credible accident could conceivably lead to some deposition of radionuclides. This situation would be controlled by necessary evacuation and establishment of radiation zones to define the boundaries of the contamination, followed by cleanup and/or immobilization of the contamination.

Recovery operations within the PRCF cell would be accomplished by flooding the cell with light water to cool and shield the remains of the incident. This would permit removal of the cover blocks so that the cause of the incident could be corrected and the damaged material removed by manually controlled underwater manipulations.

d. Fire

In the PRCF cell combustible materials have been held to a minimum. Thus the likelihood of a fire in the cell is minimized and the extent is limited. Fire from a nonnuclear event should not create a nuclear hazard. Dry chemical fire extinguishers are provided in both the reactor cell and the operating area.

A nuclear event which initiated a fire probably would be associated with the handling of fuel elements with pyrophoric cores, such as metallic uranium fuel elements. Such an incident might occur if the coolant for a preirradiated, metallic uranium fuel element were lost; however, no such material is now contemplated. If a fire of this type were to start it would burn out unchecked in the PRCF cell. The consequences of this event would be essentially the same as those described for the maximum credible accident in the Final Safeguards Analysis.

SUPPLEMENT II: ANALYSIS OF LIGHT WATER MODERATION

By

R. E. Peterson, W. K. Winegardner, and N. G. Wittenbrock

A. INTRODUCTION

The PRCF, a very low power experimental reactor, was built to supplement the Plutonium Recycle Test Reactor in the Plutonium Recycle Program. The PRCF is used for determination of basic nuclear constants of heterogeneous reactors recycling plutonium. Experiments performed in the PRCF to date have been limited to heavy water moderated cores; however, since the Plutonium Recycle Program requires nuclear data for other types of reactors, tests with light water moderation are planned. The need for light water data was recognized when the PRCF was designed, and the future conversion of the facility for light water moderation was mentioned in HW-69168.

The initial light water critical tests will be in support of a light water power reactor test of plutonium fuel planned for the EBWR at Argonne National Laboratory, where a significant fraction of the EBWR core will be loaded with mixed oxide, PuO_2-UO_2 fuel elements. The mixed oxide fuel elements for the EBWR, which will be fabricated by Hanford Laboratories, will be used in a series of light water moderated critical tests in the PRCF before the fuel is shipped to Argonne National Laboratory.

This report presents the safeguards analyses which demonstrate that the PRCF converted for light water moderation can be operated safely with EBWR, PuO_2-UO_2, fuel elements. Proposed limits are also presented for performing the critical tests using these fuel elements. Before light water moderated critical tests employing other types of fuel elements, including preirradiated fuel elements, are performed, additional analyses will be submitted for review.

B. SUMMARY

The PRCF is being modified to permit operation as a light water moderated critical facility. Data from light water critical experiments will supply physics information needed for studying the application of plutonium recycle to light water moderated reactors.

Most of the equipment installed for operation of the PRCF as a heavy water moderated facility will be used in the light water moderated experiments. The principal changes will be made to accommodate the close packed core required for a light water moderated reactor. A support structure will be installed in the existing reactor tank which will permit testing of cores up to 30 in. in diameter, and 4 ft high. Fuel rod alignment will be maintained by grid plates spaced approximately 1 ft apart vertically. A set of grid plates will be provided for each different lattice spacing.

Three control rods and three reflector safety sheets will be provided. The drive mechanisms for these control and safety devices will generally conform to the safety rod drives used in the existing heavy water moderated core. The active portion of each control rod will consist of four separate rods, having an upper cadmium poisoned section and a lower fuel follower section. Each reflector safety sheet will be a sheet of cadmium 0.030 in. thick by 8 in. wide by 4 ft long encased in plastic. The total strength of the six rods and sheets is calculated to be at least 40 mk.

Possible nuclear excursions were evaluated for critical experiments with PuO_2-UO_2 fuel to be used in the EBWR. It was found that operation of the facility in accordance with the proposed operating limits presented in the following paragraphs would not lead to a nuclear accident in which fuel melting could occur. Therefore, none of the accidents studied resulted in the release of fission products from the fuel elements. It was concluded that the severity of the credible accidents in the PRCF critical experiments for the EBWR loading should be less than that of the maximum credible accident described in the original safeguards analysis. Operation of the facility will be limited to tests with EBWR mixed oxide fuel elements, moderator-to-fuel ratios which give zero or negative void coefficients, and nominal core heights of 4 ft.

C. LIMITS

1. General

The PRCF will be operated under the limitations and restrictions set forth in this section (C. LIMITS). Information presented in the other sections of this report is to be regarded as descriptive and explanatory.

2. Operating Variables

The PRCF will be operated within the limits and in accordance with the restrictions listed in Table 2.1. Operation of the facility under these limits will be restricted to tests with EBWR mixed-oxide fuel elements, moderator-to-fuel ratios which give zero or negative moderator void coefficients, and nominal core heights of 4 ft.

TABLE 2.1

TABLE OF OPERATING LIMITS

Safety Circuit Trip Settings

Trip Function	Setting
High Neutron Flux Level	Settings equivalent to 150 W level, maximum
Reactor Period	10 sec., minimum
Seismoscope (PRTR)	
High Sensitivity	II, maximum (Intensity of Modified Mercalli Scale of 1931)
Low Sensitivity	V, maximum (Intensity of Modified Mercalli Scale of 1931)
Exhaust Air Activity	5×10^{-2} $\mu Ci/cm^3$, maximum
Aqueous Effluent Activity	5 $\mu Ci/cm^3$, maximum
Control Rods: Safety Sheets	
Insertion Time[a]	1.6 sec., maximum[b]
At-the-Ready Worth	
During Reactor Operation	32 mk, minimum[b, c]
During Fuel Changes	16 mk, minimum[b, d, e]
Rod Interlocks	Permit withdrawal of only one rod at a time
Moderator	
Minimum Critical Moderator Level (Available Reactivity Addition by Level Increase)[f]	15 mk, maximum[b]
Interlocks for Pump Operation, Moderator Addition Pumps	At least three control rods or safety sheets cocked and three inserted.

Reactivity Addition	
Ramp Addition	
More than 5 $ subcritical	5 ¢/sec/$ subcritical, maximum
Less than 5 $ subcritical	10 ¢/sec, maximum
Step Addition	
Maximum	25 ¢ (b)
Neutron Flux Monitoring During Reactivity Addition and Operation	
Safety Circuit Level Channels	Three, minimum
Safety Circuit Period Channels	One, minimum
Inherent Reactivity Effects	
Prompt Temperature Coefficient	Negative upon increasing temperature(b)
Moderator Density Coefficient	0 or negative upon decreasing density(b)
Excess Reactivity	
Maximum	1 $(g)
Power Level	100 W, maximum
Core Changes	
Control Rod-Safety Sheet Position During Core Changes	At least three cocked
Minimum Shutdown Margin	10 mk, or twice the net increase in reactivity, whichever is greater
New Cores	Incremental loading procedures will be used
Deactivation	
Minimum Requirements	(1) Reactor shut down (2) All control rods, shim rods, and safety sheets inserted with power sources locked out. (3) Repair, maintenance, or experimental work on systems covered by Operating Limits prohibited. (4) Access doors leading to control area locked.

TABLE 2.1 (contd.)

Notes:

(a) Time from signal initiation to time full reactivity worth is inserted

(b) New or revised limit

(c) Held in at least five mechanisms

(d) Calculated for the complete, critical core configuration. The safety sheets will be moved as cores are assembled to assure that sheets are installed in effective positions in the core being studied.

(e) Held in at least three mechanisms

(f) A level such that a maximum of 15 mk would be added if the moderator level were raised from the critical level to the maximum achievable level.

(g) The excess reactivity available from the console, except for that held in moderator temperature, will be limited to 1 $. Excess reactivity available from the console is defined as the excess reactivity with all control rods and safety sheets fully withdrawn and with the maximum moderator level achievable with the mechanical weir stop.

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3. Safety Circuit

The safety circuit trip functions, number of channels, number of trips required for scram, and by pass switches are listed in Table 2.2.

TABLE 2. 2

SAFETY CIRCUIT

Trip Function	Number of Channels	Trips for Scram	Bypass Switch
High Neutron Flux Level			
Start-up Channel (log count rate)	1	1	yes[(a)]
Logarithmic Channels	2	1	yes[(b)]
Linear Channels	2	1	yes[(b)]
Reactor Period			
Startup Channel (log count rate)	1	1	yes[(a)]
Logarithmic Channels	2	1	yes[(b)]
Seismoscope[(c, d)]	3	2	no
Exhaust Air Activity[(c, e)]	3	2	yes-PRTR
Aqueous Effluent Activity[(c, e)]	3	2	yes-PRTR
Removal of Reactor Cell[(f)] Access Cover Blocks	--	1	no
Transfer Lock Door Open	2	1	no

Notes:

(a) May be bypassed after a logarithmic channel is on scale

(b) One of four combinations may be bypassed at the discretion of the operator. A bypass switch with the following five positions is used:

Position 1	Bypass Linear Channel No. 1 High Level
Position 2	Bypass Linear Channel No. 2 High Level
Position 3	Unbypassed
Position 4	Bypass Log Channel No. 1 Period and High Level
Position 5	Bypass Log Channel No. 2 Period and High Level

(c) PRCF safety circuit actuated by PRTR safety circuit relays. PRTR trips are triplicated and coincident circuitry is used; trip of two of three sensing elements is required to initiate scram.

(d) Seismoscope consists of one high sensitivity and two low sensitivity pendulums.

(e) Initiates automatic confinement of PRCF; aqueous effluent activity trip also initiates automatic aqueous containment of PRCF

(f) Removal of any of the access cover blocks will automatically insert one control rod and two reflector safety sheets or two control rods and one reflector safety sheet. All other safety circuit trips, unless bypassed, initiate automatic insertion of all control rods and reflector safety sheets.

D. DESCRIPTION OF MODIFICATIONS

The building and the heavy water moderated reactor have been described in detail in HW-69168. Most of the basic reactor equipment will be used in the modification for light water moderation. The description given here deals primarily with the changes necessary for conduct of light-water-moderated critical experiments with unirradiated fuel. All modifications are designed to be easily restored to the original condition for heavy water moderation.

1. Reactor Arrangement

The existing reactor vessel, an aluminum tank 6 ft in diameter and 9 ft high, will be used for the light water moderated core. Since light water moderated experiments were contemplated at the time of the original design, the top and bottom grid plates were made in two pieces, an outer ring and a hexagonal center plate, 35 in. across the flats. The light water moderated core will be assembled to fit the hexagonal central hole. The reactor assembly is shown in Figure 2. 1.

Cores with a maximum diameter of 30 in. and maximum height of 4 ft will be accommodated. The cores will be positioned near the top of the reactor tank by a core support structure. This structure will be supported by the existing outer top grid plate. Grid plates, spaced approximately 1 ft apart vertically, will maintain fuel alignment. A set of grid plates will be provided for each different lattice spacing. The bottom plate of the core support structure will support the fuel rods; consequently, it will not be drilled for the lattice spacing. However, the bottom plate will be drilled for passage of fuel followers on the control rods and will contain slots for the passage of reflector safety sheets.

The core structure will be designed for a total live load of 4000 lb. The core will be supported in tension by six vertical aluminum rods which are fastened to a heavy aluminum ring at the top and the bottom. The top ring will be bolted to the existing outer top grid plate. The bottom ring will engage the bottom outer grid plate to prevent lateral movement. The six support rods will extend to within 1/2 in. of the bottom of the core tank.

Aluminum sleeves around each support rod, extending from the bottom ring to the core support plate, will position the aluminum core support plate approximately 4 ft above the bottom ring. Above the aluminum support plate, plastic spacer sleeves around the rods will position the lattice gird plates at approximately 1 ft intervals.

A transparent, plastic covered enclosure, approximately 4 ft high, will be centrally mounted atop the existing vessel. Support of the enclosure and alignment with the core support structure will be by bolting at the inner edge of the existing outer grid plate. The purposes of the enclosure are to provide support for the top works of the rods, atmosphere confinement, and access to the core for hand loading of fuel rods.

2. Fuel Elements

The fuel elements which will be used in the EBWR tests are shown in Figure 2.2. These elements will have mixed-oxide, PuO_2-UO_2, cores with a nominal diameter of 3/8 in. and an active length of 4 ft. Plutonium containing 8.0 at. % Pu^{240} will be used in these elements. The cores will contain 2.5% plutonium in UO_2 depleted to 0.22 at. % U^{235}. The Zircaloy-2 cladding will be nominally 0.025 in. thick.

3. Moderator System

The existing moderator system will remain essentially intact, with several important additions. A new moderator removal pump, with a pumping capacity of approximately 35 gal/min, will be installed to provide an additional means of pumping moderator from the storage tank under the reactor to the auxiliary storage tank outside of the cell. This pump will have no safety circuit or rod position interlocks on its control circuit, since the piping arrangement is such that it can pump only to auxiliary storage.

A new, vertical, 4 in. line including a quick opening butterfly valve will be provided connecting the bottom of the core vessel to the top of the storage tank. In addition to its function as a drain valve, this valve can be used as a secondary, manual shutdown device by permitting rapid lowering of the moderator level. Valve control and "open" and "shut" indicators will be mounted at the operating console. A valve opening switch also will

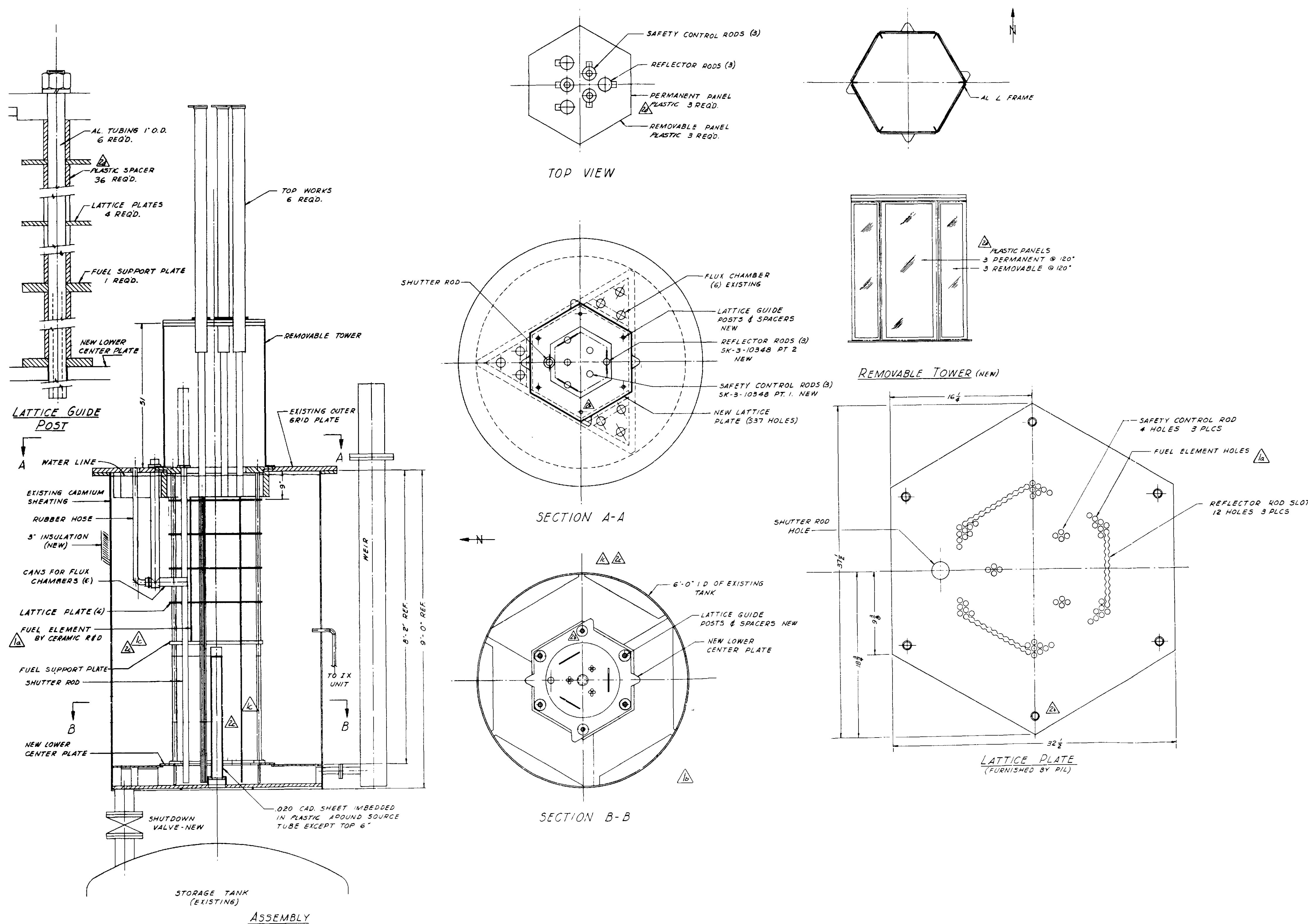

FIGURE 2.1
Reactor Assembly

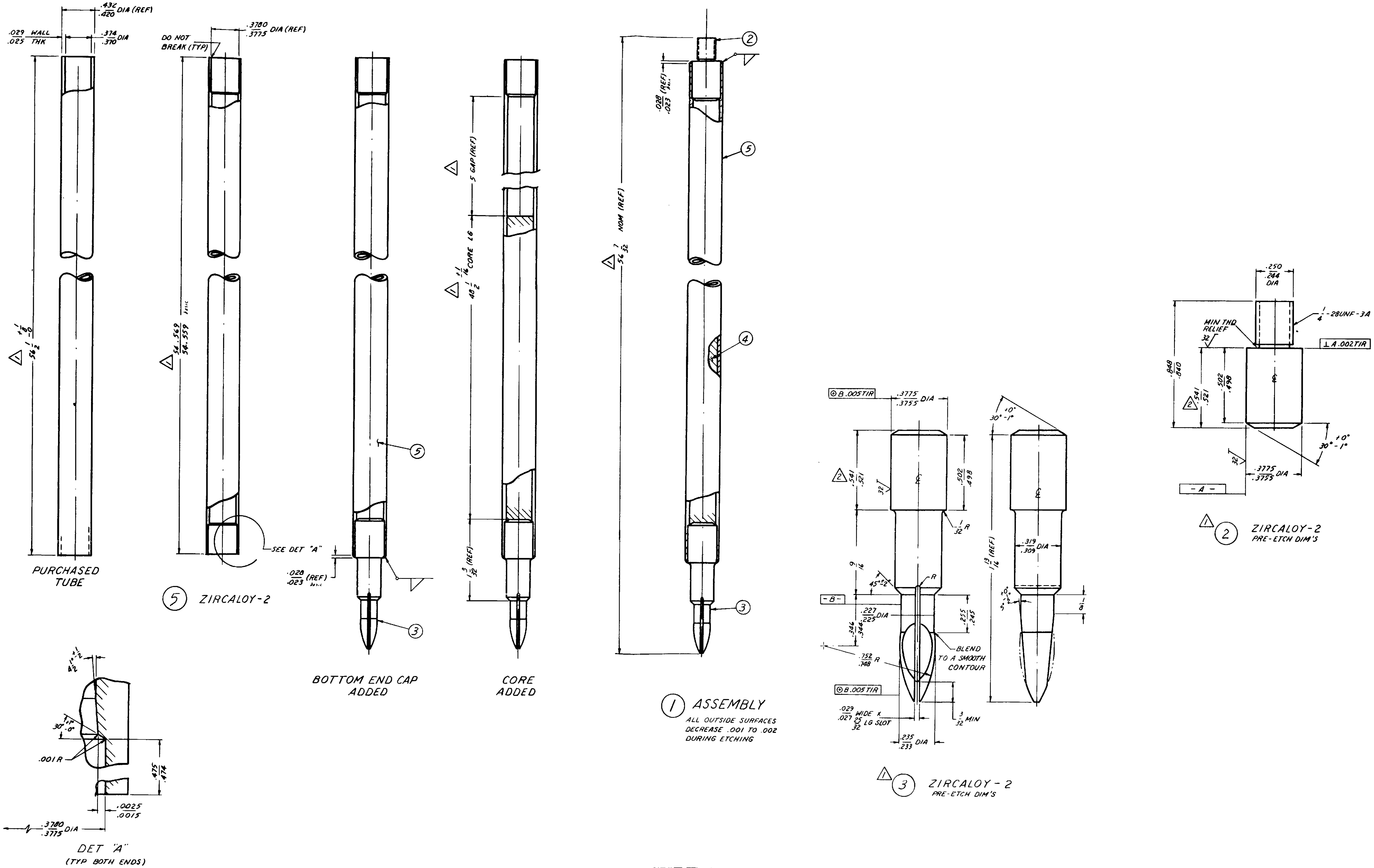

FIGURE 2. 2

EBWR Fuel Rod

be located conveniently in the cell. At the maximum rate (core vessel full) the moderator level will be reduced by approximately 0. 8 in. /sec, corresponding to a reactivity decrease rate of 1. 5 mk/sec at the top of the fuel. The existing vent line to the top of the core vessel will be enlarged to accommodate the rapid volume change.

Purity of the moderator will be maintained by recirculating a small continuous by-pass stream through an ion exchange column. The clean-up stream will flow from the reactor tank through the ion exchange column and to the moderator storage tank at a flow rate of about 1 gal/min.

The experimental program may explore the effect of moderator temperature on reactivity over the range from ambient up to 95 C. An existing moderator heater will be used to increase the temperature, and the reactor core vessel and moderator storage tank will be insulated to reduce heat losses to the cell. At the conclusion of an elevated temperature test, the moderator will be cooled either by heat transfer to the cell atmosphere or by draining the moderator from the core tank and cooling it rapidly in the storage tanks.

A schematic diagram of the moderator system is shown in Figure 2. 3.

4. Control and Instrumentation

a. Control and Safety Rods

Six rods will be provided for the core—three control rods and three reflector safety sheets. The drive mechanisms for the control rods and safety sheets will be identical, generally conforming to the safety rod drives used in the heavy water moderated core. The rods are raised and lowered by a lead screw driven by a reversible electric motor. The lead-screw ball nut is attached to an electromagnet, which holds the rod when energized and releases it when deenergized, dropping the rod into the core by gravity. The control rod and safety sheet assemblies are shown in Figure 2. 4.

The active portion of each control rod will consist of a cluster of four separate rods of the same diameter as the fuel rods in the core. The four separate rods pass through four lattice positions in the core. Each rod element will consist of an upper cadmium-poisoned section and a lower fuel follower section, both clad in Zircaloy-2. The core of the fuel follower section will contain the same fuel as the fuel rods surrounding it. In the withdrawn or "out" position, the fuel follower will be drawn into the core with the tip of the fuel follower extending through the bottom support plate. The reactivity worth of each control rod is estimated to be 8 mk.

The three reflector safety sheets will consist of a sheet of cadmium 0.030 in. thick by 8 in. wide by 4 ft. long encased in clear plastic. The plastic-encased sheet will be 1/4 in. thick. A solid plastic follower, long enough to extend through the lower support plate when the safety sheet is withdrawn, will be integrally connected to the bottom end of the safety sheet. The safety sheet will be weighted at the lower end to assure rapid insertion when it is dropped. The estimated reactivity worth of each reflector safety sheet is 6 mk. Three sets of slots will be provided in the lattice grid and support plates so that the reflector sheets may be positioned immediately adjacent to the fringe fuel rods.

The rods are designed to drop into the fully inserted position within a total elapsed time of 1.6 sec from the time of the safety circuit channel trip. A seven-position selector switch on the console permits the withdrawal of only one rod at a time. The maximum withdrawal speed is 17 in/min. The maximum reactivity insertion rates are 0.09 mk/sec for the control rods and 0.07 mk/sec for the reflector safety sheets.

In those experiments requiring a finer precision of control rod position read-out than is obtainable from the control rods, a shim control rod will be installed in the reflector zone, where it will have a reactivity worth of about 1 mk. One of the existing shutter-type control rods will be used for this purpose. This rod will be actuated by the reactor safety system.

b. <u>Safety System</u>

Nuclear instrumentation will remain as presently provided, except that the existing six chambers will be "canned" and mounted in a manner to permit adjusting their position relative to the core.

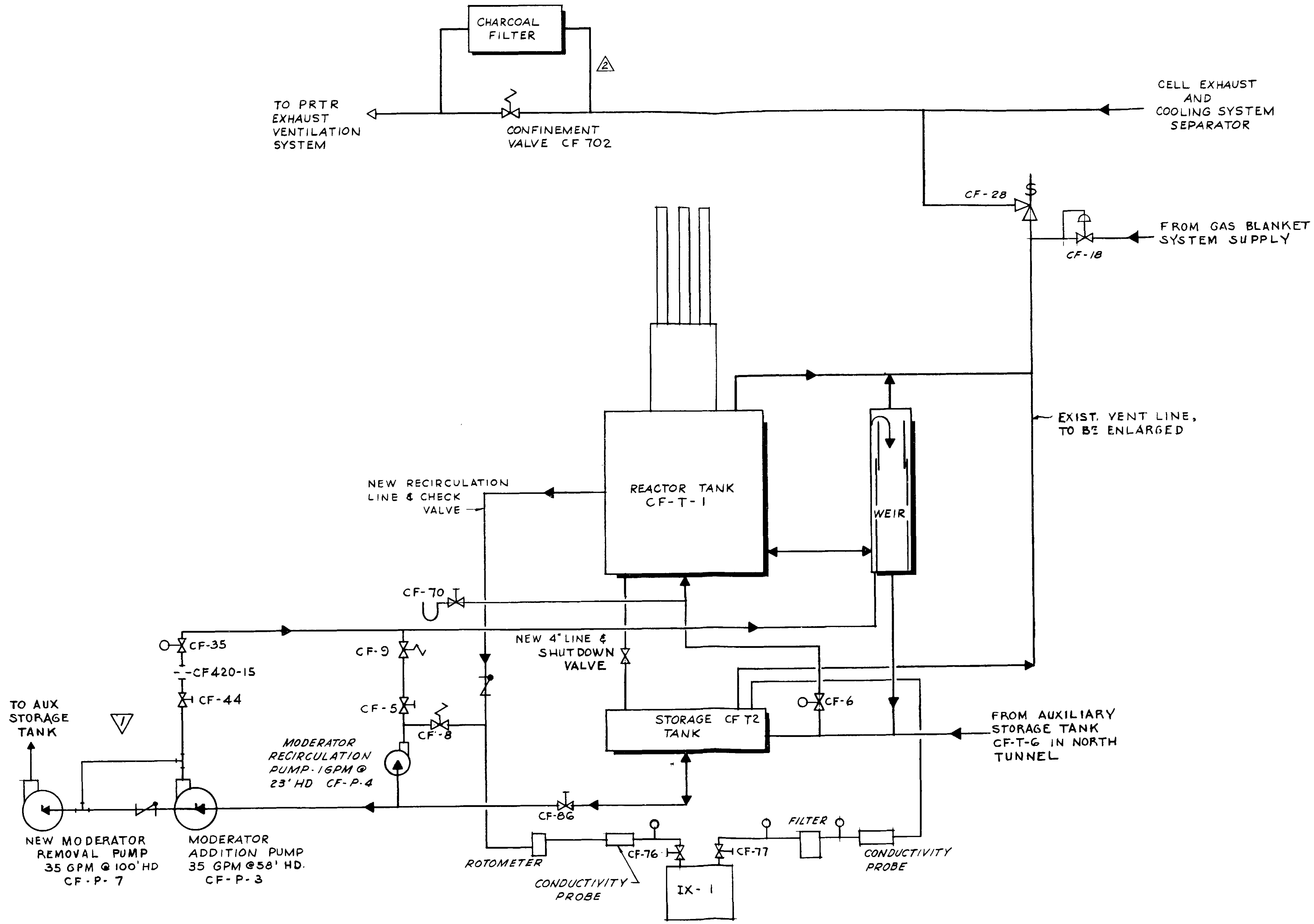

FIGURE 2. 3

Moderator Flow Diagram

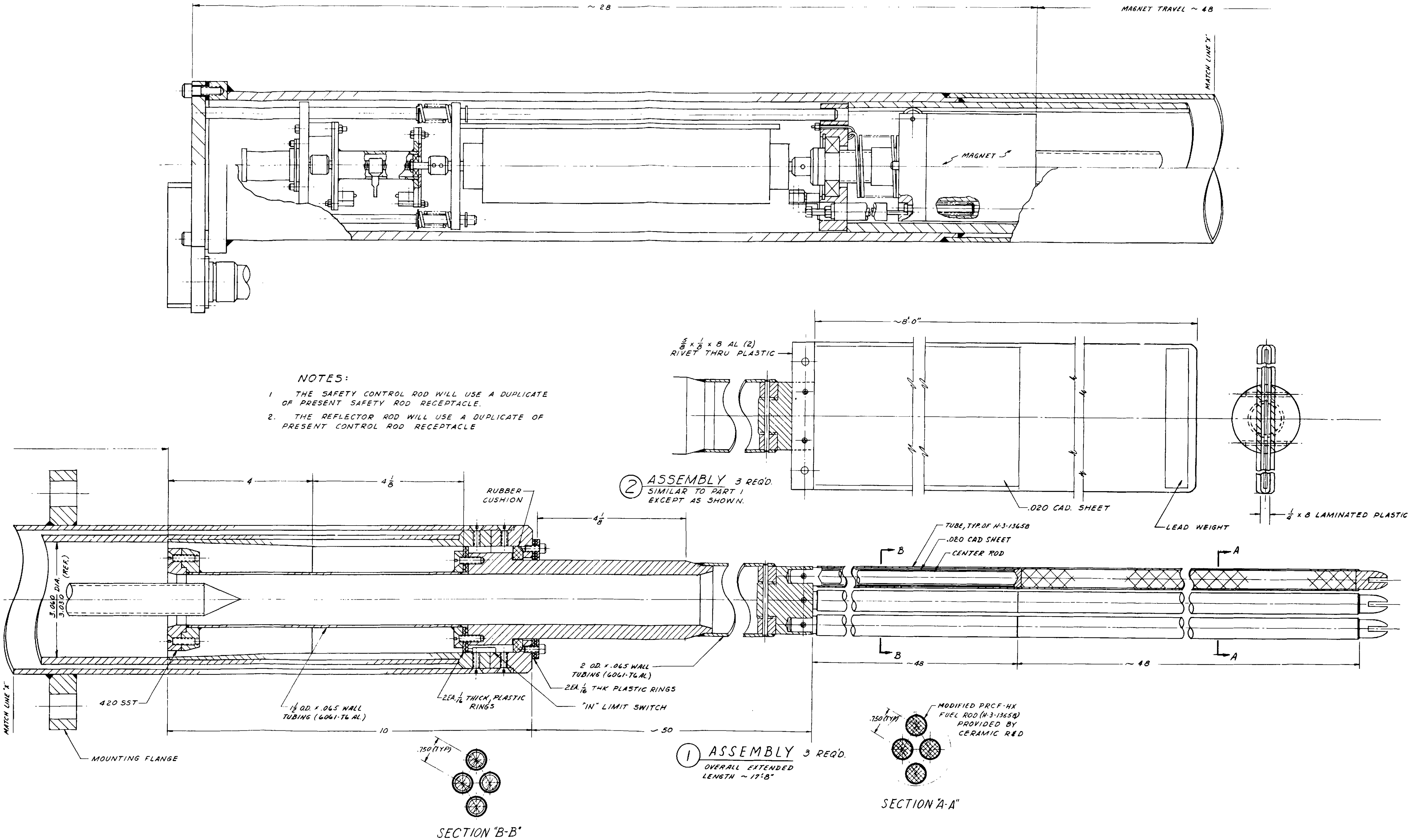

FIGURE 2. 4

Control Rod and Reflector Safety Sheet Assembly

The safety circuit will be modified by the addition of a selector switch that will permit interchange of the rods that are dropped when a cell cover block is removed. Either the "A" circuit or the "B" circuit will be opened. The "A" circuit will contain one control rod and two reflector safety sheets and the "B" circuit will contain two control rods and one reflector safety sheet. Since the reactivity worth of the rods and safety sheets is approximately equal, for the complete core loading, this arrangement will provide adequate cocked and inserted rod strength and will permit fuel changes immediately adjacent to a reflector safety sheet which must be withdrawn for fuel charging because of space limitations.

It should be noted that the safety circuit trip function which tripped the safety circuit when the linear channels were on-scale and more than one safety rod was inserted will be deleted from the safety circuit. Assurance that reactor start up would not continue without adequate cocked safety strength is given by an interlock preventing moderator addition unless three control rods or safety sheets are cocked.

A safety circuit trip has been added to actuate the safety circuit when either door of the fuel transfer lock is opened. This trip was added to insure that the rods would be inserted prior to the time a potential exists for rapid cell flooding through the lock.

The revised safety circuit is shown in Figure 2. 5.

c. Instrumentation

One of the control rods will be provided with a digital position readout unit to improve rod repositioning accuracy. The other two control rods will use the present position readout which employs a potentiometer and voltmeter. All control rods and reflector safety sheets will be equipped with full "in" and full "out" position indicators displayed at the console. The reflector safety sheets will not have intermediate positions displayed.

Reactor core temperature monitoring will be by the existing sensors and readout instruments, except that existing centrally located sensors will be removed because of space limitations in the core.

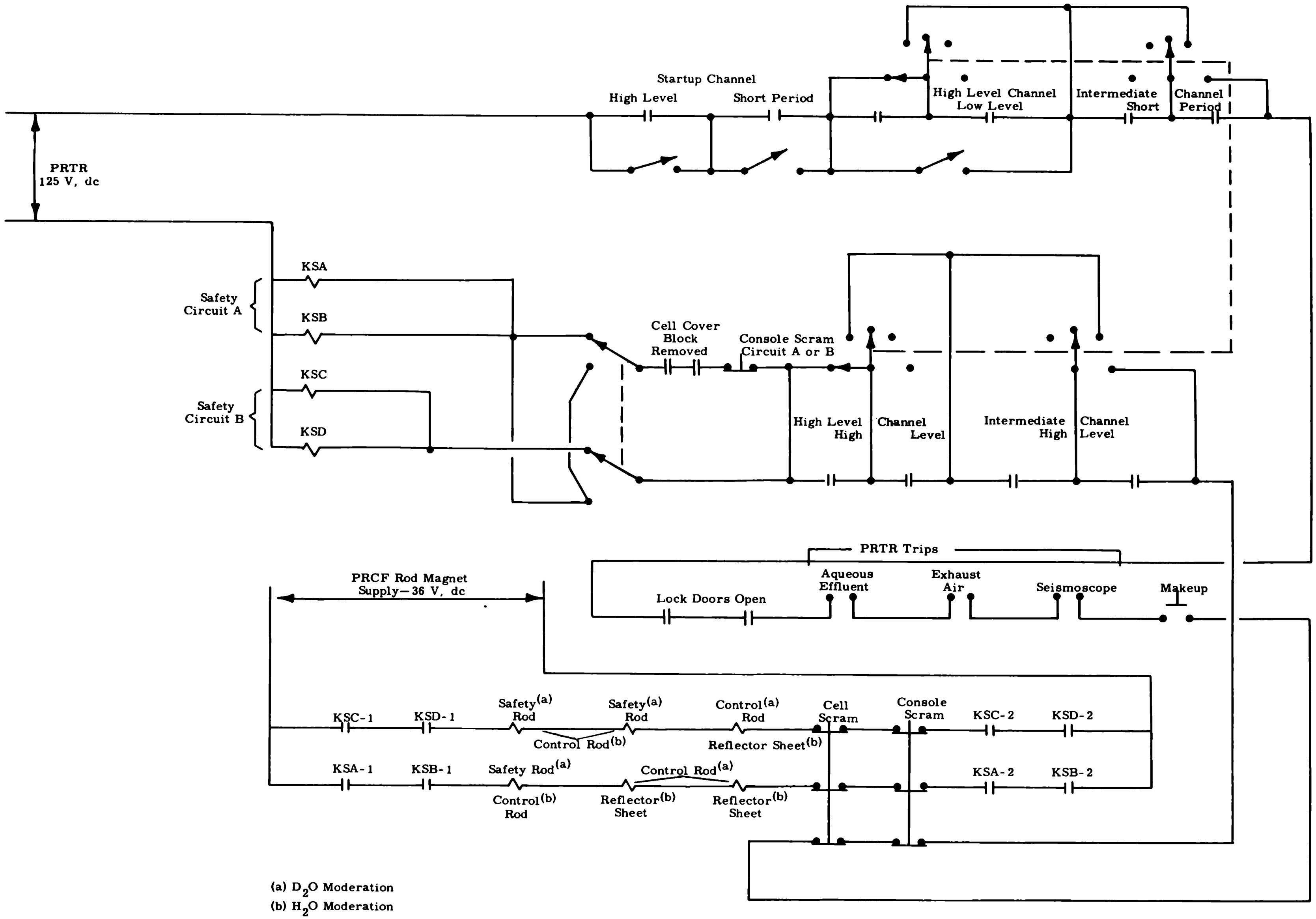

FIGURE 2.5

Revised Safety Circuit

The neutron source positioner will be modified to assure positioning of the source close to the bottom of the core.

E. OPERATION OF FACILITY

The organization for operation and the responsibilities of various Hanford Laboratories components with regard to performance, control, and review of PRCF experiments were described in HW-69168. Experiments performed on the light water moderated reactor will be similar to those described in HW-69168. Experiments will be conducted with the reactor sub-critical or with the reactor operating at power levels up to 100 W. In general, reactor operation will consist of taking neutron multiplication data and period data to obtain measurements which will assist in planning fuel loadings for plutonium fueled reactors. Typical experiments will include critical mass studies for various fuel-to-moderator ratios, measurements of basic physics parameters and reactivity coefficients, and experiments designed to determine the reactivity worth of structural materials and individual fuel elements in the various core configurations.

F. SAFETY ANALYSIS

1. Physics Characteristics

Conversion of the PRCF to light water moderator from heavy water results in substantial differences in physics parameters which form the bases for safeguards analyses. This is due essentially to the higher neutron absorption cross section and slowing down power of hydrogen compared with deuterium A direct consequence of these increased parameters is that optimum lattices are obtained at much smaller moderator to fuel volume ratios than with heavy water. From a kinetic standpoint, these lattices tend to respond more rapidly to changes in reactivity and do not have the benefit of delayed photo-neutrons, as is the case with heavy water. On the other hand, the smaller moderator-to-fuel volume ratio results in more rapid moderator temperature effects, which for undermoderated lattices can be a significant inherent safety mechanism.

The safety analyses, and hence physics parameters, were limited to a specific range of plutonium enriched loadings to cover experiments planned for EBWR type fuel only.[1] These are oxide rods 0. 372 in. OD, clad with 0. 025 in. Zircaloy-2, and 48 in. long. A fuel density of 9. 873 g/cm^3 was assumed throughout. Two enrichments of 2. 0 and 2. 5 wt% PuO_2 in fuel were selected for the calculations. Depleted uranium containing 0. 2 at. % U^{235} in uranium and plutonium with 8. 0 at. % Pu^{240} were assumed. The moderator-to-fuel volume ratio was varied for each enrichment from one to four in the calculations.

a. Critical Loadings

One-dimensional diffusion theory calculations with three neutron energy groups were used to estimate critical loadings and to derive many of the parameters needed for excursion analyses.[2] Cross sections for the two epithermal groups from 10 MeV to 500 keV and from 500 keV to 0. 532 eV were obtained from GAM-1 calculations.[3] Thermal group data were computed from the TEMPEST code.[4]

Although this coarse group structure may not yield precise calculations of critical loadings, the parameters obtained (prompt lifetime, perturbation coefficients, etc.), are adequate for excursion analyses.

The estimated critical loadings for EBWR fuel in the PRCF are shown in Figure 2. 6. The range of experimental interest will cover those critical loadings achievable with approximately 1000 rods or less and moderator to fuel ratios less than 4.

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(1) L. C. Schmid, Letter to N. G. Wittenbrock. Information: PRCF—Light Water Experiments. January 14, 1963.

(2) J. J. Regimbal. Unpublished data.

(3) G. D. Joansu and J. S. Dudek. A Consistent P_1 Multigroup Code for the Calculation of the Fast Neutron Spectrum and Multigroup Constants, GA-1850. June, 1961.

(4) R. H. Shudde and J. Dyer. TEMPEST, A Neutron Thermalization Code, Atomics International, NAA, Program Description. September, 1960.

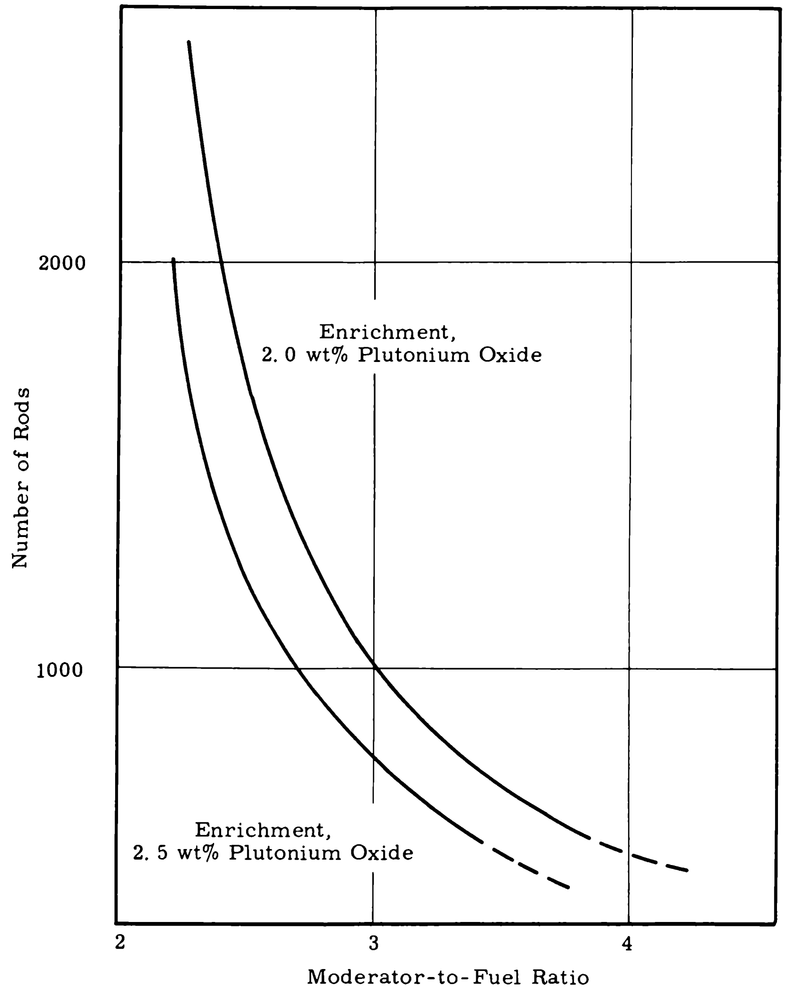

FIGURE 2.6

Estimated Critical Loadings: EBWR Fuel in PRCF

b. Doppler Coefficient

Very little of the U^{238} is displaced by plutonium in EBWR fuel, so that the fuel temperature coefficient of reactivity should remain close to that measured for natural uranium oxide rods. The temperature dependent resonance integral RI(T) is given by

$$RI(T) = RI(T_o)\left[1 + \beta(T^{\frac{1}{2}} - T_o^{\frac{1}{2}})\right]$$

where for UO_2, $\beta = (0.58 + 0.5\ S/M)\ 10^{-2}$

and $RI(T_o) = 4.15 + 26.6\ (S/M)^{\frac{1}{2}}$, where $T_o = 293$ K.

The effect of the Pu^{240} was also included by assuming a self-shielding factor equivalent to that for U^{238}. The Pu^{240} contribution to the total resonance integral is only about 6%.

The reactivity change accompanying a fuel temperature change from T_o to T can be shown to be[1]

$$\Delta k/k = \beta (\ln p_o)\ (T^{\frac{1}{2}} - T_o^{\frac{1}{2}})$$

where p_o is the resonance escape probability for the lattice at temperature T_o. The quantity $\beta(\ln p_o)$ is given in Table 2. 3 as a function of moderator-to-fuel ratio and enrichment for EBWR fuel.

TABLE 2. 3

FUEL TEMPERATURE COEFFICIENT

	$\beta\ (\ln p_o)$		
Enrichment, wt%	M/F = 2.0	M/F = 3.0	M/F = 4.0
2.0	-1.39×10^{-3}	-9.30×10^{-4}	-6.97×10^{-4}
2.5	-1.41×10^{-3}	-9.39×10^{-4}	-7.04×10^{-4}

These coefficients are of the same order as those obtained for D_2O lattices of mixed oxide fuel, previously analyzed.[1]

c. Moderator Coefficient

In addition to the fuel temperature coefficient for these loadings, which responds almost instantaneously to limit an excursion, the moderator void coefficient can also be a relatively fast inherent shutdown mechanism for lattices which are undermoderated. This is caused by the relatively small volume of water present in the lattice to absorb heat transferred from the fuel, resulting in a rapid temperature rise. The moderator coefficient of reactivity was obtained from a perturbation calculation in which an importance-weighted value for moderator density change was obtained for each loading. The moderator void coefficient becomes less negative as the moderator-to-fuel ratio is increased from an undermoderated value, until the coefficient

(1) R. E. Peterson. Nuclear Parameters - PRTR Mixed Oxide Fuel Safeguards Analysis, HW-74346. July 18, 1962.

becomes positive. This occurs at a ratio of about 4 for the fuel types investigated. A positive moderator coefficient tends to reinforce an excursion; thus, experimental lattices for which this occurs will be avoided, although the Doppler coefficient may compensate to some extent.

d. Delayed Neutron Fraction

Delayed neutrons from Pu^{239} fission are less abundant than from U^{235} fission by about a factor of 3. Since the mixed oxide fuel for the EBWR will be depleted in U^{235}, the number of delayed neutrons from this source will be relatively small. However, the delayed neutron yields characteristic of the fraction of U^{235} fissions were weighted with those from Pu^{239} to calculate the values given in Table 2.4 for six groups.[1]

TABLE 2.4

DELAYED NEUTRON FRACTION

	Fractional Yield, β_i	Decay Constant, λ_i, sec^{-1}
(1)	0.0791×10^{-3}	0.0128
(2)	0.644×10^{-3}	0.0301
(3)	0.484×10^{-3}	0.124
(4)	0.777×10^{-3}	0.325
(5)	0.207×10^{-3}	1.12
(6)	0.102×10^{-3}	2.69
Total	$2.31 \times 10^{-3} = \beta$	

Fast fissions in U^{238} were assumed to be negligible. The total delayed fraction β, is somewhat lower than for similar mixed oxide fuels in the D_2O lattice (HW-69168) previously studied because of the lower fraction of U^{235} fissions and the loss of the photoneutron contribution.

e. Prompt Neutron Lifetime

The prompt neutron lifetime, 1^*, in the lattices studied is only about 10% of that characteristic of the heavy water moderated PRCF. This is due primarily to the higher neutron absorption rate in the light water lattice.

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(1) Reactor Physics Constants, ANL-5800.

Values were calculated from the general multigroup formula

$$1^* = \frac{\int \sum_i \frac{\varphi_i^+(\bar{r})\ \varphi_i(\bar{r})}{V_i}\ d\bar{r}}{\int \sum_i \sum_j X_j\, \varphi_j^+(\bar{r})\ \nu\, \Sigma_{fi}\, \varphi_i(\bar{r})\ d\bar{r}}$$

where φ_i and φ_i^+ are the group flux and adjoint flux respectively,

ν is the neutron yield per fission

X_j is the normalized fission spectrum

Σ_{fi} is the group fission cross section

V_i is the group velocity.

Integration over the reactor and reflector in three energy groups yielded values of from 2.9×10^{-5} to 4.5×10^{-5} sec. Response to reactivity changes will thus be somewhat faster than is presently experienced in the heavy water moderated PRCF.

f. Other Characteristics

In addition to the parameters described, which are constants for a particular loading, estimates of reactivity addition capability due to fuel rod drop-in and increasing moderator level were made. Fuel rod drop-in calculations were carried out using essentially the same perturbation technique which was used to derive the moderator density coefficient. When fuel is added to a vacant lattice position, an equal volume of moderator is displaced. Thus, although the addition of fuel by itself contributes positively to the reactivity, the contribution of the loss of moderator from a given location may be negative or positive, depending upon the degree of moderation. Also, the importance of both fuel and moderator varies as a function of radial position in the core. Calculation of perturbation coefficients for both fuel and moderator were made for the loadings investigated. The maximum net reactivity addition estimated from these coefficients was less than $1.

The moderator level coefficient of reactivity was assumed to vary according to the one dimensional bucking formula

$$\frac{1}{k}\frac{dk}{dh} = \frac{2\pi^2 M^2}{h^3}$$

where M^2 is the lattice migration area and h is the moderator level. The total reactivity addition between two levels H_1 and H_2 is then

$$\Delta k/k = \pi^2 M^2 \left[\frac{1}{H_1^{\,2}} - \frac{1}{H_2^{\,2}}\right].$$

The calculated reactivity addition on raising moderator at a 35 gal/min rate is shown in Figure 2. 7.

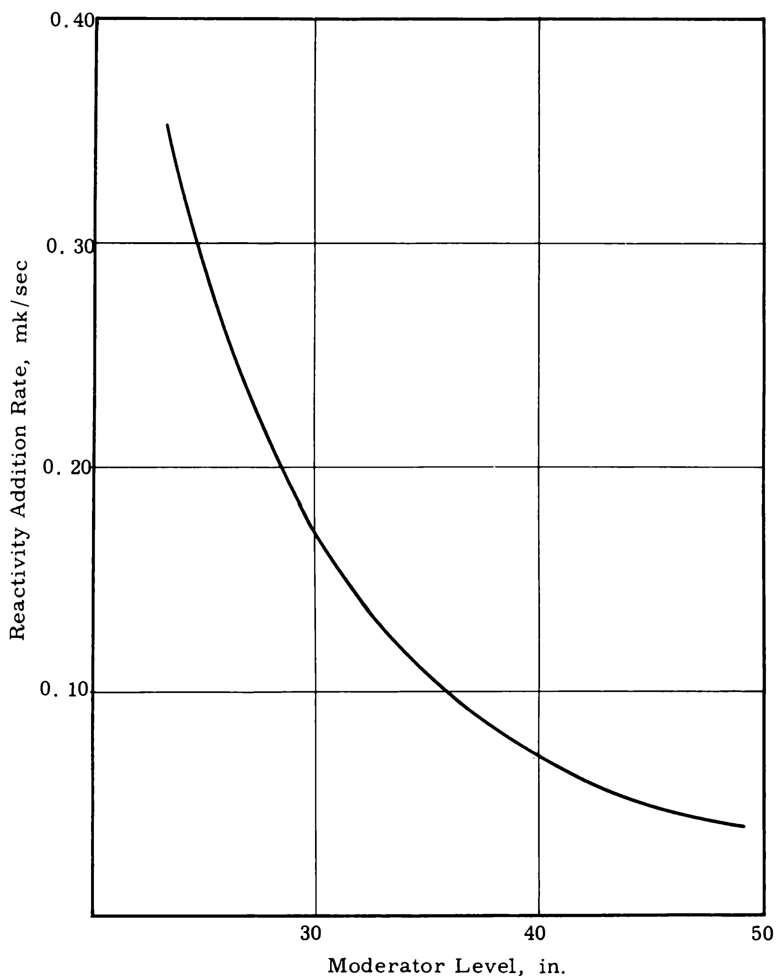

FIGURE 2. 7

Reactivity Addition Rate Upon Increasing Moderator Level at 35 gal/min

2. Failures

a. Mechanical Failures

(1) Stuck Rod

The effect of one control rod or reflector safety sheet sticking and failing to drop into the core upon safety circuit trip is a reduction of the reactivity worth of the safety system by an amount equal to the strength of the individual rod. The estimated total reactivity worth of the safety system is 40 mk and the worth of the strongest rod is 10 mk, leaving a net strength of 30 mk if the strongest rod were stuck. Reactor accident analyses presented in Section F. 3. a indicate that an excursion initiated by continuous withdrawal of a control rod would be terminated before core damage could occur even if the safety system strength were reduced to 20 mk by rod failures.

(2) Fuel Transfer Lock Door Failure

The fuel transfer lock between the PRCF cell and the PRTR loadout canal presents the potential for a cell flooding accident. However, full insertion of the reflector safety sheets and the control rods before cell flooding occurs would hold the reactor subcritical.

Design test data show that failure of the lock-door pneumatic seals would allow water to flow into the cell at a rate of 45 gal/min and that the capacity of the cell sump pump is greater than this flow rate. If the sump pump should fail, it would take about 4 hr to fill the cell to a level which would allow water to flow into the reactor tank from the top. However, cell flooding would not result in criticality, since an infinite radial reflector is provided for all cores and the minimum critical moderator level permitted is such that the safety system can hold the reactor at least 10 mk subcritical with the reactor tank completely filled with water.

Rapid flooding of the cell could occur only if the interlocks preventing simultaneous opening of both lock doors were to fail. The following interlocks are provided in the hydraulic system for operating the lock doors.

- The cell door cannot be opened unless hydraulic pressure is applied to the canal door closing piston and a float actuated valve indicates that the lock is dry.

- The canal door cannot be opened unless hydraulic pressure is applied to the cell door latching mechanism and a float actuated valve indicates that the lock is full of water. The canal door, which opens outward into the canal, would be held closed by the pressure of the water in the canal whenever the lock is dry or the cell door is open. Therefore, the potential for rapidly flooding the cell exists only when the canal door is open. The safety circuit trip provided to drop all of the rods and safety sheets before either lock door is opened will ensure that the control rods and safety sheets are inserted whenever the potential exists for both lock doors to be opened at the same time.

b. Electronic Failure

The neutron flux monitoring instruments are the only electronic instruments which could affect reactor safety upon failure. Fail-safe design and/or duplication of channels protects against safety circuit instrument failure. A trip of any one of four neutron flux level channels, two intermediate level logarithmic channels and two linear high level channels, will trip the safety circuit. A bypass selector switch permits by-passing of only one of the four channels at a time. This also assures that at least one of the two logarithmic period channels is unbypassed at all times. Low ion chamber voltage is annunciated at the control console. As shown in Figure 2.5, the main safety circuit relays have double pole contacts which break the rod magnet circuit on both sides of the magnets.

c. Procedural Errors

The design and operating characteristics of critical facilities are such that procedural errors present possibilities for nuclear excursions. Several interlocks were included in the original design of the facility to prevent operator error. Most of the interlocks provided initially are of value for reactor startup and operation with either light water or heavy water. These features are summarized below:

- Interlocks are provided to prevent movement of the control rods in the direction of increasing reactivity unless the startup channel or one of the intermediate channels is on-scale.

- The seven-position rod selector switch prevents applying power to more than one control rod, the shim rod, or one reflector safety sheet at a time. Drive switches for these rods will be spring loaded, requiring the operator to have his hand on the switch in order to move a rod in the direction of increasing reactivity.
- Simultaneous moderator addition, with either of the two moderator addition pumps, and control rod or reflector safety sheet withdrawal are prevented by interlocks. The system is interlocked such that the pumps will not operate unless three control rods or safety sheets are fully withdrawn and the other three are fully inserted. Only one of the two moderator addition pumps can be operated at a time.
- An audible monitor, furnished in conjunction with the scaler, provides an audible indication of increasing subcritical flux levels. The monitor has speakers in both the operating area and the reactor cell to warn of increasing subcritical flux levels.
- A portion of the safety circuit is automatically tripped when the reactor cell cover block for any access opening is removed. This will prevent entrance into the reactor cell when the reactor is operating.
- Interlocks are provided such that only one of the fuel-transfer-lock doors can be opened at a time. The hydraulic control system for the lock is arranged so that:
  a. Three valves must be positioned before water pressure can be applied to the cylinder which opens the canal door (a valve which is positioned by a float which indicates that the lock is full, a pneumatic actuated valve which indicates that the seal is depressurized, and a hydraulic valve which indicates that hydraulic pressure is being applied to latch the the cell door).

  b. Four valves must be positioned before water pressure can be applied to the cylinder which opens the cell door (a valve positioned by a float which indicates the lock is empty, a pneumatic valve which indicates that the seal is depressurized a hydraulic valve which indicates that pressure is being applied to close the canal side door, and a mechanically actuated valve which indicates that the cell door is unlatched.

d. Power Failure

The reactor will be shut down immediately upon loss of electric power. Loss of ac power to the rectifier which provides the dc power for the rod magnets will deenergize the magnets, allowing the rods to fall. Loss of ac power will also open the 4 in. shut-down valve, resulting in lowering the moderator level in the reactor tank to 5 ft.

As shown in Figure 2.5, the rod magnet circuit contacts of safety circuit relays KSA, KSB, KSC, and KSD are normally open; therefore, loss of dc safety circuit power will also deenergize the rod magnet circuit, allowing the rods to fall.

e. Handling of Cell Cover Blocks

The large cell cover blocks, weighing about 10 tons, are positioned using a mobile crane operating through the removable roof section. Conceivably, a block could be dropped into the cell while being moved. To prevent the possibility of achieving a critical array as the result of a dropped cover block, either the water will be removed from the reactor tank and storage tank or the fuel will be removed before a cover block, other than the small personnel access cover block, is moved.

3. Accident Analysis

a. Rod Withdrawal at the Maximum Rate

The maximum continuous reactivity addition rate by withdrawing a control rod will be 10 ¢/sec* whenever the reactor is less than 5 $ subcritical. The excess reactivity available from the console will be limited to 1 $; i.e., after the final increment of fuel is charged the excess reactivity will not

*Reactivity can also be added by three other mechanisms, reflector safety sheets, shim control rod, and moderator addition. Interlocks prevent addition of reactivity with more than one of these mechanisms at a time. The maximum permissible rate of reactivity addition by any one of these mechanisms is 10 ¢/sec when the reactor is less than 5 $ subcritical.

exceed 1 $ with the rods in the most reactive position and the moderator at the maximum level permitted by the mechanical weir stop.

Several different lattice spacings will be investigated in performing the critical tests with the EBWR mixed-oxide fuel. Analyses of rod withdrawal accidents are presented for two different lattices, which represent the extremes of moderator void coefficient which may be encountered in the EBWR critical experiments. Calculated physics parameters for the two cores are summarized in Table 2. 5.

TABLE 2. 5

CORE PHYSICS PARAMETERS

Triangular Lattice Spacing, in.	0. 71	0. 81
Approximate Number of Fuel Pieces	1100	400
Moderator-to-Fuel Ratio	2. 70	4. 0
Doppler Coefficient, $\Delta k/k/F$ (100 to 500 F)	-1.29×10^{-5}	-0.59×10^{-5}
Moderator Void Coefficient, $\Delta k/k/\%$	-2.64×10^{-3}	0
Moderator Temperature Coefficient, $\Delta k/k/F$ (70 to 200 F)	-6.85×10^{-5}	0
β	2.31×10^{-3}	2.31×10^{-3}
l^*	2.9×10^{-5}	4.5×10^{-5}

Reactor excursions initiated by continuous withdrawal of a control rod were simulated on an analog computer. Continuous rod withdrawal was started either with the reactor subcritical ($k_{eff} = 0.98$) or with the reactor operating at a power level of 100 W. Excursions were terminated by either a safety circuit trip or the inherent shutdown mechanisms, Doppler coefficient and moderator void coefficient. Details of the analog simulation of these excursions are given in Appendix C. The basic assumptions used in all of the accidents analyzed are:

- Reactivity was added continuously at a rate of 10 ¢/sec until either the safety circuit tripped or a total of 1 $ of excess reactivity had been added.
- The safety circuit period trips failed.
- For excursions terminated by scram the safety circuit tripped at a power level of 150 W. The negative reactivity insertion versus time after scram is shown in Figure 2. 8.

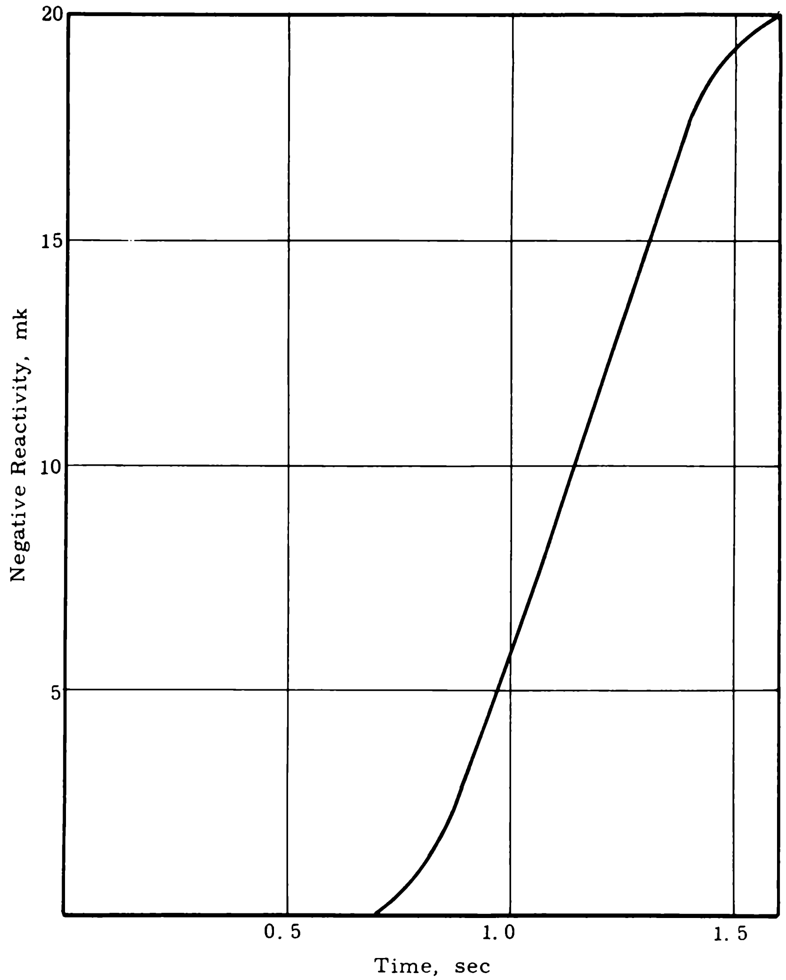

FIGURE 2. 8

Negative Reactivity Insertion After Safety Circuit Trip

- For excursions terminated by inherent shutdown mechanisms the Doppler coefficient and the moderator void coefficient were the mechanisms used.

(1) Safety Circuit Trip at 150 W

The power level transient and the integrated energy release for the accidents with the reactor initially 20 mk subcritical are shown in Figure 2. 9. Sufficient reactivity was added prior to the safety circuit trip to achieve prompt criticality. No perceptible fuel temperature rise was observed.

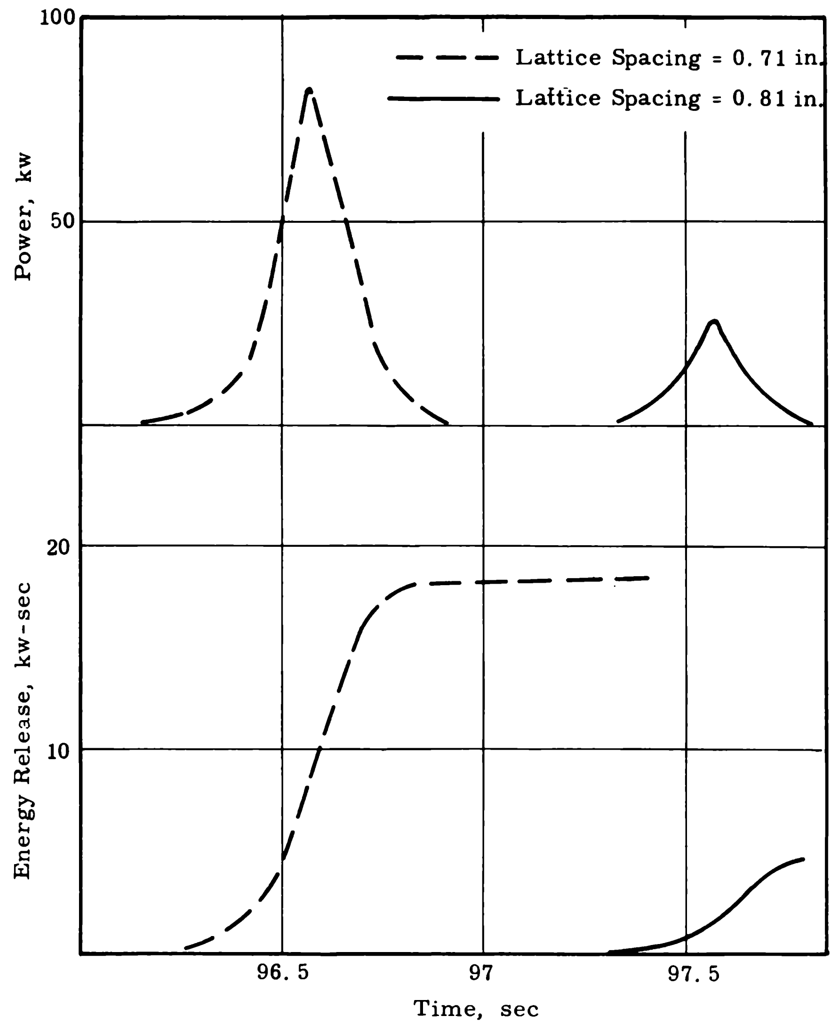

FIGURE 2. 9

Continuous Rod Withdrawal Accident from Subcritical, Safety Circuit Trip at 150 W

The response to a reactivity addition rate of 10 ¢/sec when the reactor power level was 100 W was almost identical for the two cores. The maximum power level achieved was only slightly higher than the trip point of 150 W, and again there was no perceptible increase in fuel temperature. The power level transient and integrated energy release for these accidents are shown in Figure 2. 10.

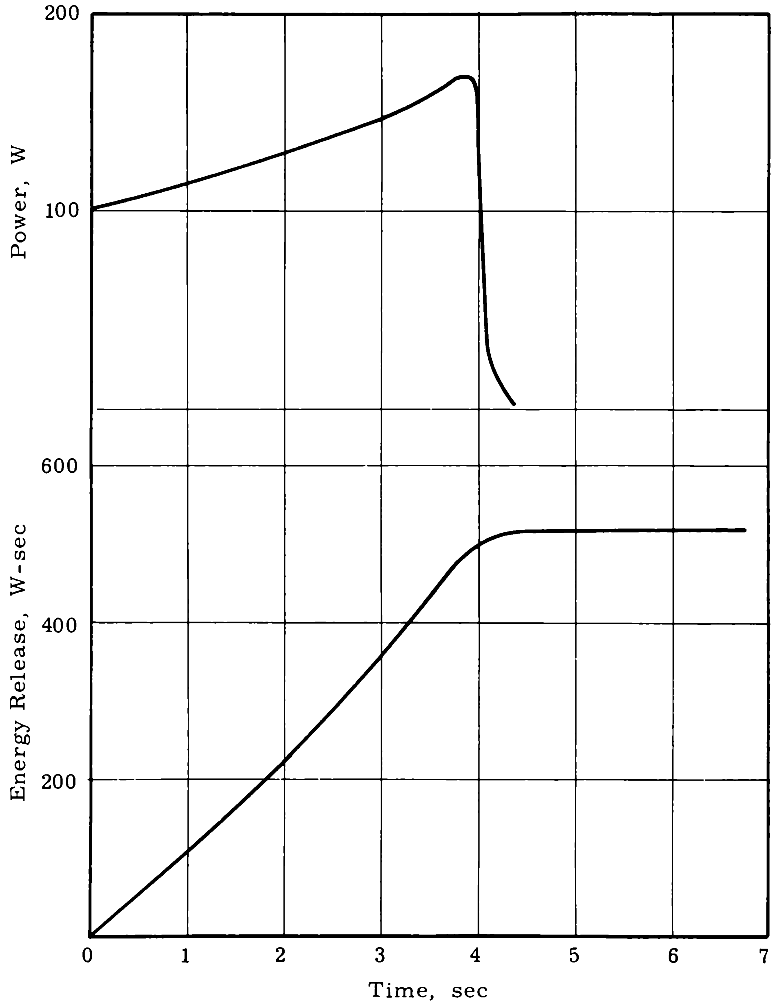

FIGURE 2. 10

Continuous Rod Withdrawal Accident from 100 W Power Level, Safety Circuit Trip at 150 W

(2) Safety Circuit Fails to Trip

The power level transient, integrated energy release, and maximum fuel core temperature are shown in Figure 2. 11 for the accidents resulting from continuous addition of reactivity at 10 ¢/sec until 1 $ of excess reactivity had been inserted. The analysis showed that for the core with the strong negative void coefficient the reactor would be shut down about 2 sec after the peak power was achieved, but for the core with the zero void coefficient the power level would begin to increase again after the first power peak was passed. However, rapid vaporization of moderator would expel water from the core and shut down the reactor.

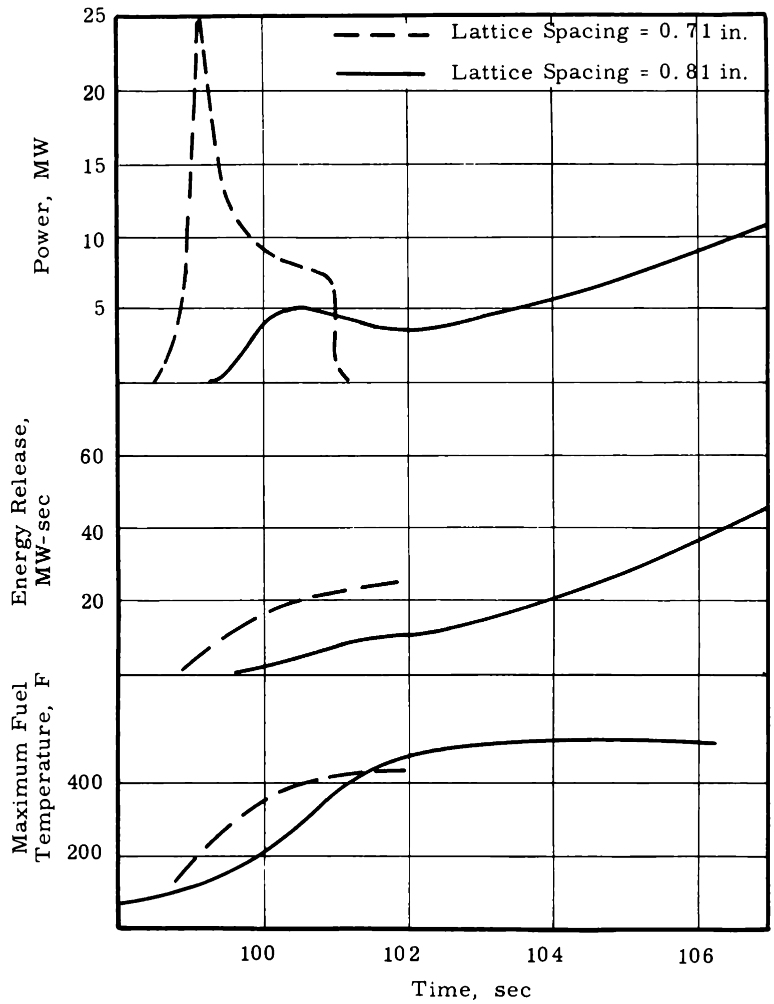

FIGURE 2.11

Continuous Rod Withdrawal Accident from Subcritical, Safety Circuit Fails to Trip

The results of the excursions initiated from a power level of 100 W by continuous withdrawal of a rod are shown in Figure 2.12. As in the accidents just described, the excursion in the core with the strong moderator void coefficient would be terminated by the moderator void a few seconds after the peak power is reached. The excursion in the core with the zero void coefficient would be terminated by expulsion of moderator when rapid moderator vaporization starts about 50 sec after the first power peak.

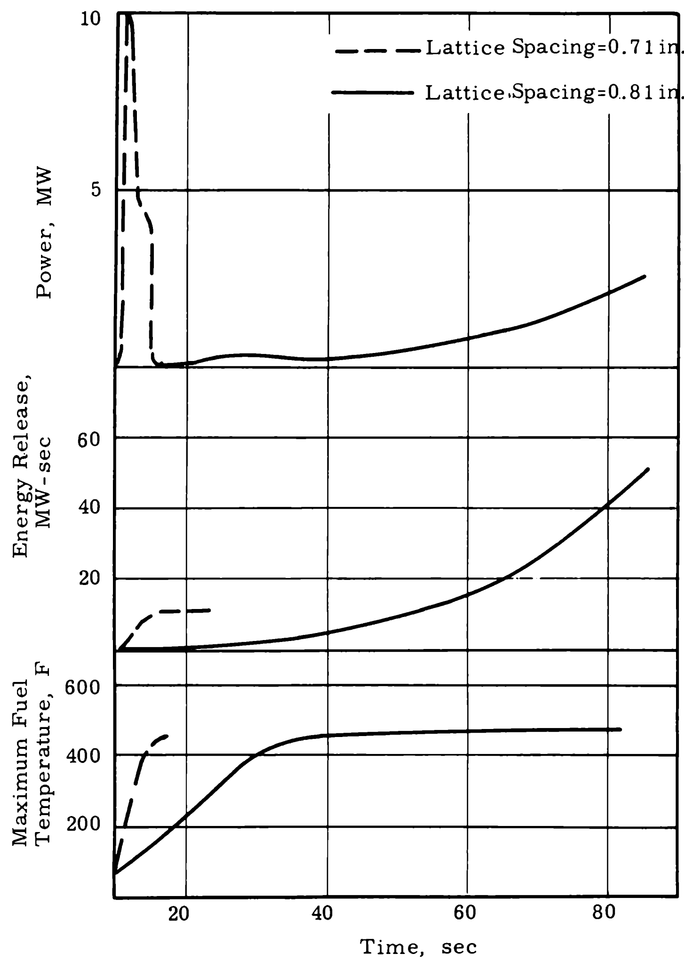

FIGURE 2. 12

Continuous Rod Withdrawal Accident from 100 W Power Level, Safety Circuit Fails to Trip

In none of these accidents did the fuel temperature reach a point which would lead to fuel element failure. Therefore, there should be no release of fission products from the fuel elements.

b. Rod Shoot Out

The only conceivable mechanism for rapid removal of rods from the core is the build-up of a high pressure in the reactor tank. A pressure buildup in the reactor tank would be relieved through the rod support cage mounted on top of the reactor. Three of the side panels will be designed to act as blowout panels in the event of a pressure build-up in the reactor tank. These panels will blow out at a pressure of approximately 1 in. of water.

c. Cold Water Accident

The moderator will be heated during experiments where it is desired to determine the value of the overall temperature coefficient. At the conclusion of such tests the moderator will be cooled by one of two methods:

- Heat loss from the core vessel to the cell atmosphere
- Controlled reduction of temperature.

When cooling is to be achieved by heat loss to the cell atmosphere, the maximum cooling rate by free air convection will be less than 1 F° /hr, equivalent to a reactivity addition rate of about 0.083 mk/hr. The calculated overall temperature coefficient for a core loading with a large void coefficient is 3.6 ¢/F° (70 to 200 F). Continuous cooling at a rate of 2.8 F°/sec would be required to postulate accidents similar to those described in Section F. 3. a., where it was assumed that reactivity was added at a rate of 10 ¢/sec. Continuous cooling rates of this magnitude cannot be achieved by natural convection from the insulated tank, even if water were rising around the tank because both of the lock door seals and the sump pump had failed.

Controlled reduction of the moderator temperature while the moderator is in the reactor core will not be possible, because the moderator must be drained from the core for rapid cooling. In accordance with the limits in Section C. (LIMITS), the core loading will be adjusted, when necessary, to give a maximum of 1 $ excess reactivity available from the console at the moderator temperature for the next test. As the cooled moderator is returned to the reactor tank, multiplication measurements will be taken, and an inverse multiplication plot will be used to determine the reactivity status of the core.

d. Manual Withdrawal of Fixed Poison or Experimental Poison

Whenever the reactor tank contains fuel, manual removal of any of the rods or safety sheets will be prohibited unless all of the moderator is drained from the reactor core. No physical experimental devices have been defined for the facility which could act as poisons other than structural materials and core instrumentation. In any case, procedures will be used to assure that the reactor is initially subcritical and remains subcritical during the manual withdrawal of experimental poisons.

4. Effect on Maximum Credible Accident

None of the postulated accidents presented in this report results in the release of fission products. Therefore, the severity of the maximum credible accident will not be increased by performing light water moderated experiments utilizing EBWR fuel elements.

SUPPLEMENT III: ADDITIONAL STUDIES— LIGHT WATER MODERATION

by

W. K. Winegardner

A. INTRODUCTION

The PRCF will be modified to permit operation as a light water moderated test facility. Proposed limits and a description of the modifications required were presented in Supplement II. As stated there, initial light water tests will be in support of a light water power reactor test of plutonium fuel planned for the EBWR at Argonne National Laboratory. A recent analysis of EBWR fuel requirements indicated that the enrichment needed in the mixed-oxide, PuO_2-UO_2 fuel elements (those fabricated by Hanford Laboratories) will be 1. 5% plutonium in UO_2. The descriptive information of Supplement II indicated that the fuel used in PRCF-EBWR tests would contain 2. 5% plutonium in UO_2 and transient analyses were presented for cores with this enrichment. This report presents additional descriptive information concerning the results of transient analyses of rod withdrawal accidents which were performed for the 1. 5% case.

B. SUMMARY

The results of the transient studies indicate that the severity of a rod withdrawal accident would be less in a core loading with 1. 5% enrichment that in a core loading with 2. 5% enrichment. The severity of the credible accidents during PRCF experiments using EBWR fuel should be less than that of the maximum credible accident described in HW-69168 and Supplement I.

Information presented in this report is to be regarded as descriptive and explanatory and in no way revises the limits and restrictions previously presented in Supplement II.

C. TRANSIENT ANALYSIS

1. General

The analyses presented in this section for the 1.5% case were performed using the proposed limits of Part C of Supplement II. The change in physics parameters upon decreasing the enrichment from 2.5 to 1.5% is of insufficient magnitude to significantly affect the severity of an excursion. The calculated values (lattice spacing of 0.71 in.) of the total delayed neutron fraction, prompt neutron lifetime, and Doppler coefficient used for the transient studies of the two enrichment cases are given in Table 3.1 for comparison purposes.

TABLE 3.1

COMPARISON OF PHYSICS PARAMETERS[(a)]

PuO_2	1.5%	2.5%
Delayed Neutron Fraction	2.56×10^{-3}	2.31×10^{-3}
Prompt Neutron Lifetime, sec	5×10^{-5}	2.9×10^{-5}
Doppler Coefficient, $\Delta k/k/F$ (100 to 500 F)	-1.65×10^{-5}	-1.29×10^{-5}

(a) The value of the moderator void coefficient is strongly dependent on the moderator-to-fuel ratio of the core loading. Only core loadings with calculated negative or zero void coefficients were studied, since the proposed limits of Supplement II prohibit the use of a positive moderator void coefficient.

The transient studies consisted of the analysis of simulated reactor excursions which were initiated by a ramp reactivity addition rate of 10 ¢/sec (continuous rod withdrawal). The equations used to describe the excursion were solved on an analog computer. Details of the analog simulation are in Appendix D. The transient studies and basic assumptions used are similar to those described in Part F.3.a of Supplement II. The basic assumptions used in the performance of the studies are summarized below:

- Continuous rod withdrawal was started either with the reactor subcritical (k_{eff} = 0.98) or with the reactor operating at a power level of 100 W.

- Transients were initatied by a reactivity addition rate of 10 ¢/sec.
- The excess reactivity was limited to 1 $.
- The safety circuit period trips failed.
- For excursions terminated by scram the safety circuit tripped at a power level of 150 W. The negative reactivity insertion versus time after scram is shown in Figure 2.8.
- For excursions terminated by inherent shutdown mechanisms, the Doppler effect and moderator void (steam formation) and temperature effects were the mechanisms used.
- The fuel element surface heat transfer coefficient (boiling heat transfer coefficient) is shown in Figure C-1, Appendix C.

Calculated physics parameters for the 1.5% case are given in Table 3.2. Parameters are similar to those used in the analysis of the 2.5% case except for the number of fuel rods required for a critical core configuration. More refined calculations indicate that the core size will be smaller than was originally anticipated.

TABLE 3.2

CORE PHYSICS PARAMETERS

Triangular Lattice Spacing, in.	0.71	0.81
Number of Fuel Rods	410	350
Moderator to Fuel Ratio	2.7	4.0
Doppler Coefficient $\Delta k/k/F$ (100 to 500 F)	-1.65×10^{-5}	-1.12×10^{-5}
Moderator Void Coefficient $\Delta k/k/\%$ Void	-4×10^{-3}	-2.5×10^{-3} and 0[a]
Moderator Temperature Coefficient $\Delta k/k/F$ (70 to 200 F)	-10.4×10^{-5}	-6.5×10^{-5} and 0[a]
Delayed Neutron Fraction	2.56×10^{-3}	2.56×10^{-3}
Prompt Neutron Lifetime, sec	5×10^{-5}	5×10^{-5}

(a) Moderator reactivity coefficient calculated for this loading as indicated; transient was also performed in which it was assumed coefficient was zero.

2. Safety Circuit Trip at 150 W

The power transient and energy release during the transient for the accidents with the reactor initially 20 mk subcritical are shown in Figure 3. 1. No perceptible fuel element core temperature rise was observed.

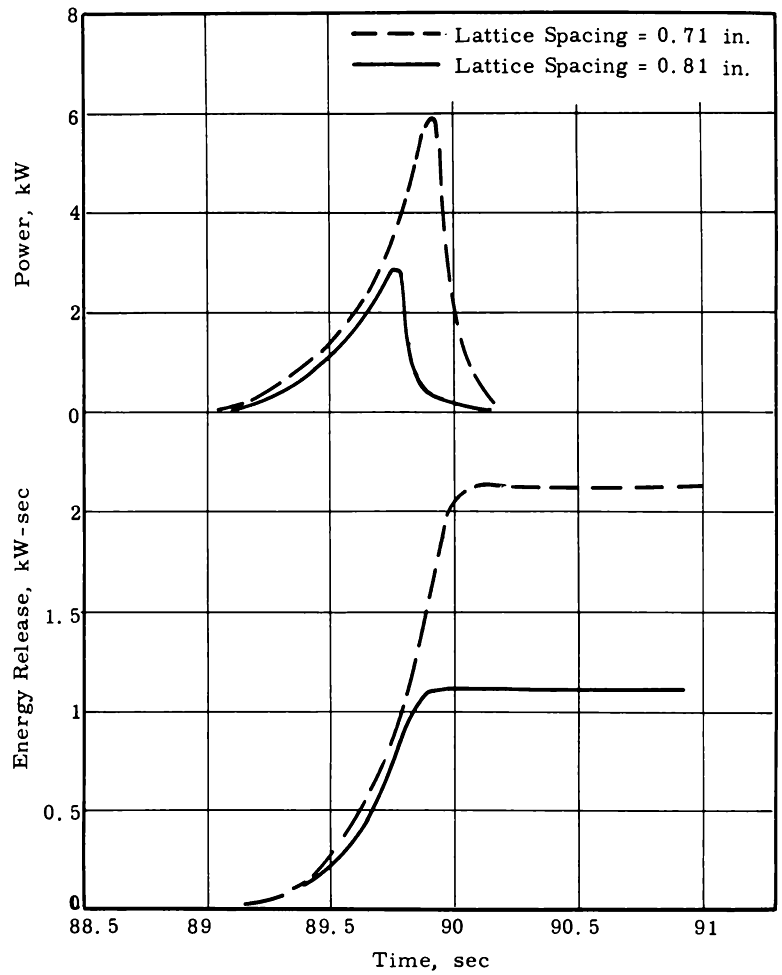

FIGURE 3. 1

Continuous Rod Withdrawal Accident from Subcritical, Safety Circuit Trip at 150 W

The response to an addition rate of 10 ¢/sec while operating at 100 W is shown in Figure 3. 2 for the core with the 0. 71 in. lattice spacing. Results for the core with 0. 81 in. lattice would be almost identical.

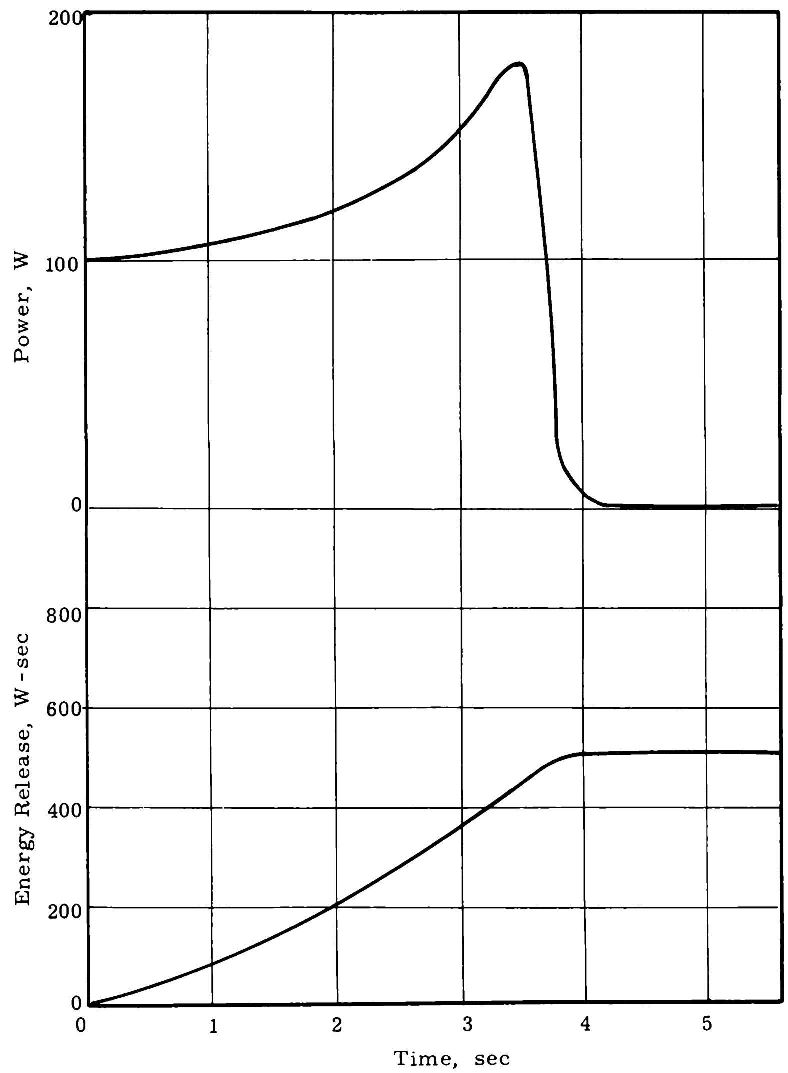

FIGURE 3.2

Continuous Rod Withdrawal Accident from 100 W Power Level, Safety Circuit Trip at 150 W

3. Safety Circuit Fails to Trip

Results obtained for the 1.5% case transient studies which were over-ridden by inherent shutdown mechanisms were almost identical. The power rise was terminated by the Doppler effect and the response after the power peak was similar for all the cases studied. This was true for cases with different nuclear parameters as well as for transients which were initiated from subcritical and while operating at 100 W. The reason for the above was the use of more refined heat transfer equations. The effect of using the equations is discussed in Appendix D. The power level transient, integrated energy release, and fuel element core and surface temperatures are shown in Figure 3. 3 for one of the subcritical studies (lattice spacing of 0. 81 in.).

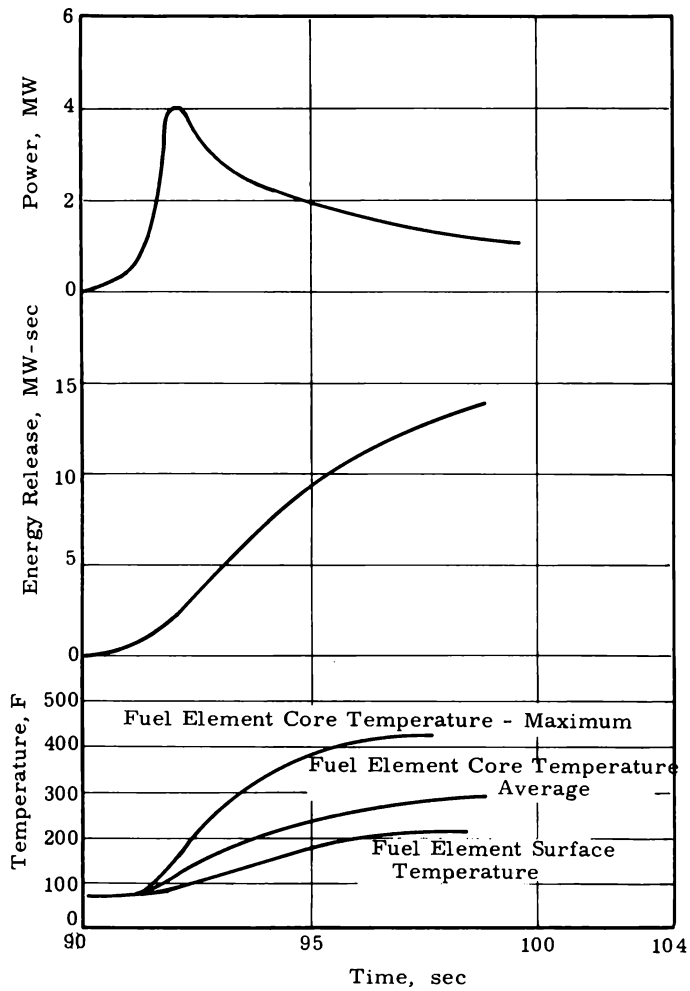

FIGURE 3. 3

Continuous Rod Withdrawal Accident From Subcritical, Safety Circuit Fails to Trip

4. Discussion of Results

The magnitude of results (peak power, energy release) obtained for the 1. 5% case was less than that obtained for the 2. 5% case. For transients overridden by inherent shutdown mechanisms, the primary reason for the difference in results presented in this document and those presented in Supplement II was the use of the more refined heat transfer equations.

In none of the analyses studied did the fuel element temperature reach a point which would lead to fuel element failure. Therefore, there should be no release of fission products from the fuel elements.

SUPPLEMENT IV: ANALYSIS OF VARIOUS LIGHT WATER MODERATED CORE LOADINGS

by

W. K Winegardner and N. G. Wittenbrock

A. INTRODUCTION

Modification of the PRCF to permit operation as a light water moderated facility was described in Supplement II. This modification will permit extension of the experimental program to light water moderated critical tests in support of the Plutonium Recycle Program.

The operating limits of Supplement II presently restrict the light water test program to critical tests using mixed oxide, PuO_2-UO_2 fuel elements fabricated by Hanford Laboratories for use in the EBWR. It was indicated in Supplement II that light water moderated critical tests using other types of fuel elements were planned and that additional analyses would be submitted for review prior to conducting these tests. This section presents safety analyses of core loadings consisting of Pu-Al fuel elements, oxide or ceramic fuel elements, and mixtures of Pu-Al and ceramic fuel elements. Proposed changes in operating limits for the facility are also presented.

B. SUMMARY

Transient studies for light water moderated core loadings consisting of Pu-Al fuel elements (1. 8 wt% and 5. 0 wt% plutonium) and a light water moderated core loading of mixed oxide (PuO_2-UO_2, 1. 5 wt% PuO_2) fuel elements were performed where it was assumed that • the transient or excursion was terminated by the safety system and • the excursion continued until overridden by inherent negative reactivity effects. The excursions were simulated on an analog computer. No perceptible fuel element temperature rise was observed when the transient was terminated by the safety system. The results of the studies where it was assumed that the safety circuit failed indicate that • the Doppler effect is the primary mechanism for initially overriding an excursion in a core loading consisting of ceramic fuel, and • excursions in a core loading consisting of metallic Pu-Al fuel and having a negative

moderator void coefficient are overridden almost immediately upon local boiling at the fuel element surface. In general, for the case of safety circuit failure, it was found that the magnitude of the initial power peak was • of the same order of magnitude for all core loadings with relatively strong inherent negative reactivity effects and • relatively independent of the enrichment level of the fuel. It was concluded that the severity of the credible accidents for light water moderated experiments should be less than that of the maximum credible accident described in HW-69168 and in Supplement I.

C. LIMITS

The PRCF will be operated under the limitations and restrictions set forth in Tables 2.1 and 2.2 of Part C, Supplement II except for the following revision to the operating limit concerning inherent reactivity effects.

Inherent Reactivity Effects	Limit
Loadings consisting entirely of Pu-Al fuel elements	Overall moderator void coefficient for the reactor will be negative and at least 0.3 mk/% void.
Other core loadings	Overall Doppler coefficient for the reactor will be negative. Overall moderator void coefficient for the reactor will be zero or negative.

Information presented in the other parts of this section is to be regarded as descriptive and explanatory.

D. REACTOR ARRANGEMENT

The reactor arrangement and the control and safety systems were described in Supplement II. Fuel elements for core loadings other than the mixed oxide-EBWR loading will be cylindrical, Zircaloy-2 clad fuel rods with a nominal unclad diameter of 1/2 in. Both PuO_2-UO_2 and Pu-Al fuel rods will be used, and core loadings may be all ceramic, all metallic, or mixed loadings of ceramic and metallic fuel. Mixed loadings will be used for tests where it is not possible to achieve criticality with all ceramic loadings of relatively low enrichment because of the size of the core or in those tests where there is a limited number of ceramic fuel elements available of a particular enrichment. Spike enrichment in the form of Pu-Al fuel elements will be added to a zone or region of the ceramic fuel in sufficient quantity to achieve a critical core configuration.

A typical reactor assembly for light water moderation is shown in Figure 2.1 of Supplement II. The fuel elements are positioned near the top of the reactor tank. One fuel rod occupies each individual lattice position. Fuel element alignment is maintained by plastic grid plates. Several sets of grid plates will be required because the test program for a given fuel element type will include lattice spacing as a variable.

The control rods and safety sheets are described in Supplement II. The active section of the control rods will be slightly different for the various fuel element types, since each rod of the four-rod cluster consists of an upper cadmium poison section and a lower fuel follower section. The fuel follower will be fabricated from the same material as the fuel elements used in the region where the rod is installed. The length of the poison sections and fuel follower sections of the control rods will be approximately the same as the length of the fuel rods in the core under test. Holes and slots will be provided in the grid plates for each core arrangement in positions such that the control rods and safety sheets can be placed in effective positions for the core size under consideration.

The safety circuit for the facility is described in Supplement II.

E. SAFETY ANALYSIS

1. General

The types of fuel elements used in the light water moderated PRCF experiments determine the values for the physics and thermal constants which form the bases for transient studies. The safety analysis presented in this report consists solely of studies to determine the effect of these different values of physics and thermal constants upon the response characteristics of typical core loadings. Since the analyses of component failures and the description of devices to protect against procedural errors given in Supplement II are applicable for core loadings consisting of the other types of fuel elements as well as for the mixed oxide, EBWR loading, no further discussion is presented.

2. Transient Studies

Reactor excursions initiated by addition of reactivity at a rate of 10 ¢/sec, the operating limit for the PRCF, were simulated on an analog computer. Studies were performed for cases where it was assumed • that the transient was terminated by the safety system (safety circuit high flux level trip) and • that the excursion continued until overridden by inherent negative reactivity effects.

Calculated physics and thermal constants for the core loadings studied are summarized in Table 4.1. The constants used for one of the mixed-oxide, EBWR studies are also included for comparison purposes. The constants for the PuO_2-UO_2 core were calculated for a core loading consisting of fuel elements containing 1.5 wt% PuO_2 in UO_2 (natural uranium).* The constants for the two Pu-Al cores were calculated for core loadings consisting of 1.8 wt% plutonium (3.7 ft core) and of 5.0 wt% plutonium (2 ft core). The calculated values of the moderator void coefficient for the above core loadings are shown as a function of moderator-to-fuel ratio in Figure 4.1.

TABLE 4.1

PHYSICS AND THERMAL CONSTANTS FOR VARIOUS FUEL LOADINGS

Parameter	Core Configuration			
	PuO_2-UO_2		Pu-Al	
	1.5 wt% PuO_2 Natural U^{235}	1.5 wt% PuO_2[(a)] Depleted U^{235}	1.8 wt%	5 wt%
Number of fuel rods	250	350	500	200
Core length, ft	3	4	3.7	2
Nominal fuel element diameter, in.	1/2	3/8	1/2	1/2
Doppler Coefficient, Δk/k/F (100 to 500 F)	-0.78×10^{-5}	-1.12×10^{-5}	0	0
Moderator Void Coefficient, Δk/k/% void	-2×10^{-4} [(b)]	-2.5×10^{-3} [(b)]	-3×10^{-4}	-3×10^{-4}
Delayed neutron fraction	3.34×10^{-3}	2.56×10^{-3}	2.1×10^{-3}	2.1×10^{-3}
Prompt neutron lifetime, sec	5.5×10^{-5}	5×10^{-5}	7×10^{-5}	7×10^{-5}
Fuel element thermal time constant, sec[(c)]	8.08	4.65	0.06	0.06

(a) EBWR fuel

(b) Results show that the magnitude of the initial power peak for the studies where it is assumed that the safety system fails is independent of this value.

(c) $RC = (1/8\pi k + 1/2\pi rh)(\pi r^2 C_p \rho)$ where r, k, C_p, ρ, and h are the radius, thermal conductivity, heat capacity, and density of the fuel element core material and the interface (fuel element-cladding material) heat transfer coefficient, respectively.

*The transient studies for the EBWR core loading presented in Supplement III were based on a core containing 1.5% PuO_2 in UO_2 depleted to 0.22 at% U^{235}.

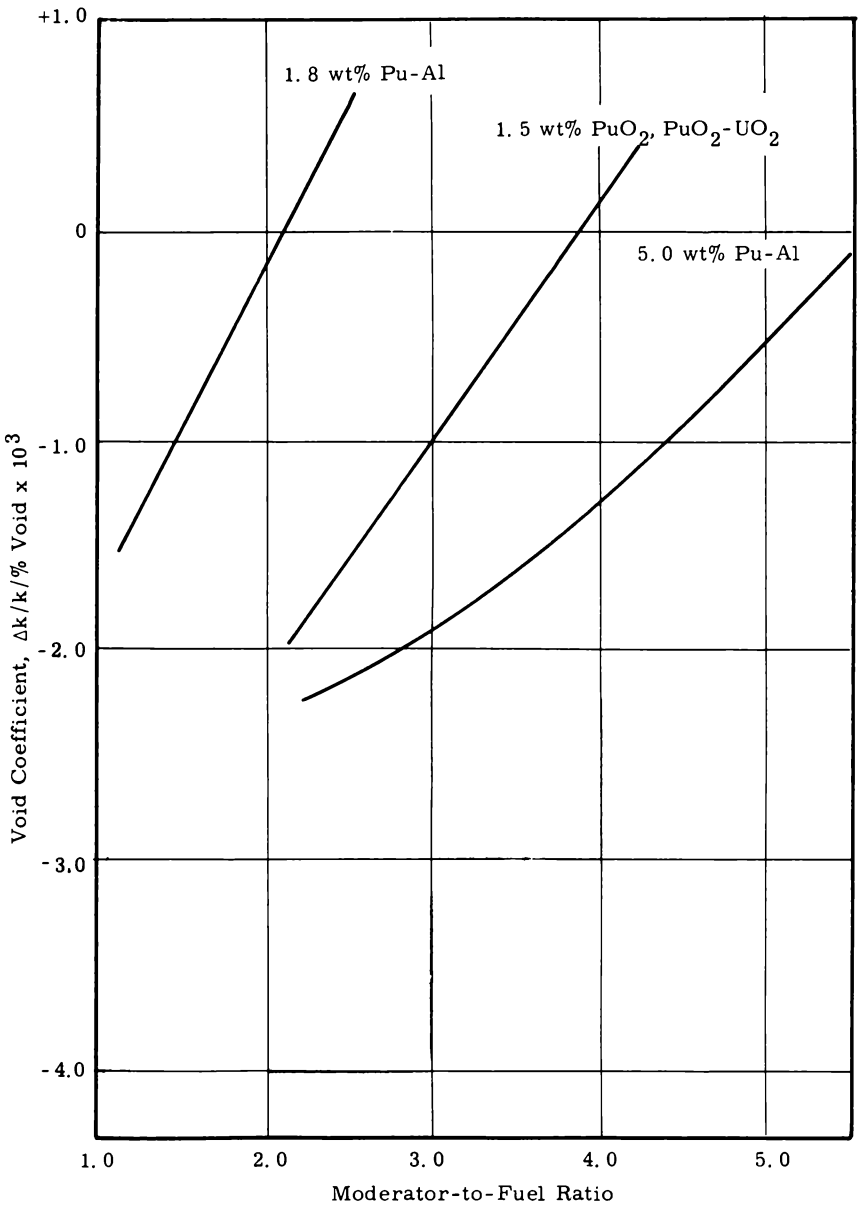

FIGURE 4.1

Void Coefficient of Core Loadings

The method of analysis used to describe the transients assumes a one energy group model for the neutron kinetics equations. Unsteady state heat transfer equations are derived for average conditions from heat balance considerations using a thermal resistance-capacitance model. The method of analysis is described in Reference (1).

(1) W. K. Winegardner. Approximate Method for Analyzing a Self-Limiting Reactor Transient, HW-77147. September 27, 1963.

Other assumptions used include:

- The safety circuit period trips fail
- Transients initiated from subcritical were initiated from $k_{eff} = 0.99$
- For excursions terminated by reactor scram the safety circuit tripped at a power level of 150 W. The negative reactivity insertion versus time after scram is shown in Figure 2.8 (20 mk worth with a total insertion time of 1.6 sec).
- For excursions terminated by inherent shutdown mechanisms, the Doppler effect and steam void effect upon film boiling at the fuel element surface are the major, prompt mechanisms for reactivity feedback. The void volume formed by boiling is obtained from an equation presented by Janssen, et al.[1] The equation includes the assumption that the void volume is determined by the thickness of the vapor film that separates the fuel element from the moderator under steady-state film boiling conditions and that the steam film forms immediately upon initiation of boiling, i.e., when the surface temperature of the fuel element reaches 212 F.
- No heat is transferred from the fuel element until the onset of boiling. The fuel element surface heat transfer coefficient used is shown in Figure C-1, Appendix C.

a. Startup Accidents

(1) Safety Circuit Trip at 150 W (Reactor Initially 10 mk Subcritical)

The power level transient and energy release for excursions initiated from a subcritical flux level in the PuO_2-UO_2 core are shown in Figure 4.2 for continuous reactivity addition and with the excess reactivity limited at 1 \$. Similar results for the two Pu-Al cores are shown in Figure 4.3. No perceptible fuel element temperature rise was observed.

It should be noted that the excess reactivity limit for the PRCF is 1 \$. However, the consequence of greater than 1 \$ available excess reactivity is illustrated by the excursions initiated by continuous addition of reactivity at 10 ¢/sec, also shown in Figures 4.2 and 4.3.

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(1) E. Janssen, W. H. Cook, and K. Hikido. Metal-Water Reactions: 1. A Method for Analyzing a Nuclear Excursion in a Water Cooled and Moderated Reactor, GEAP-3073. October 15, 1958.

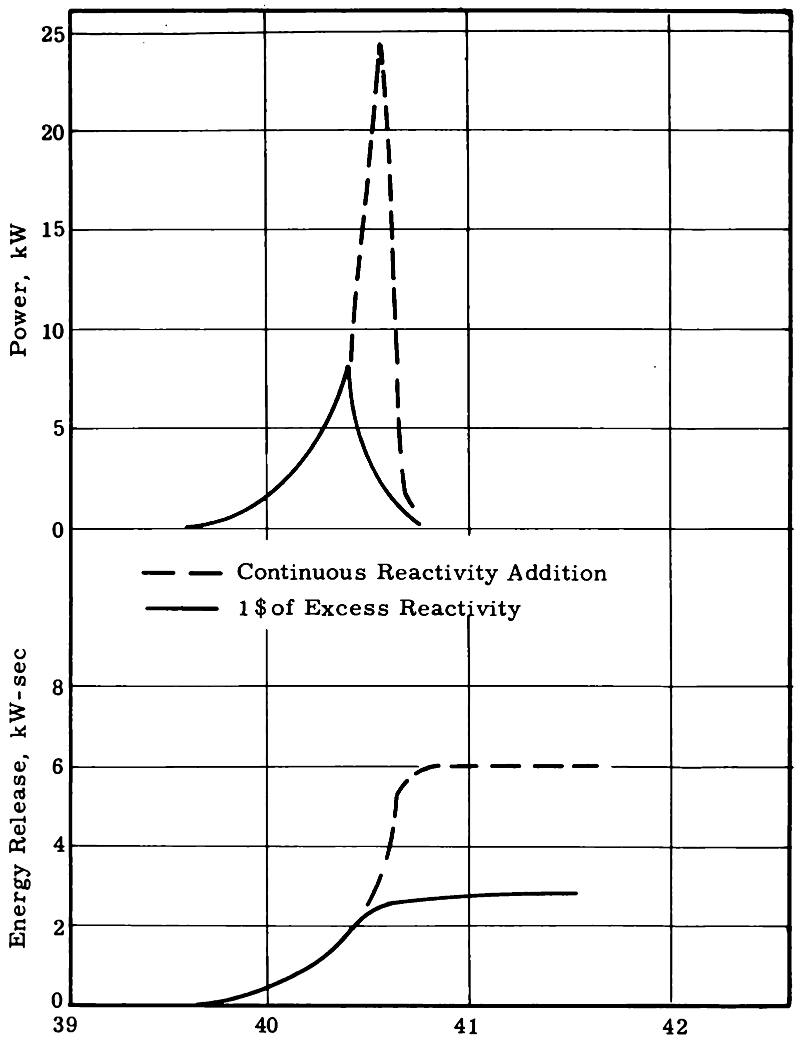

FIGURE 4. 2

PuO_2-UO_2 Core Transient,
Safety Circuit Trip at 150 W

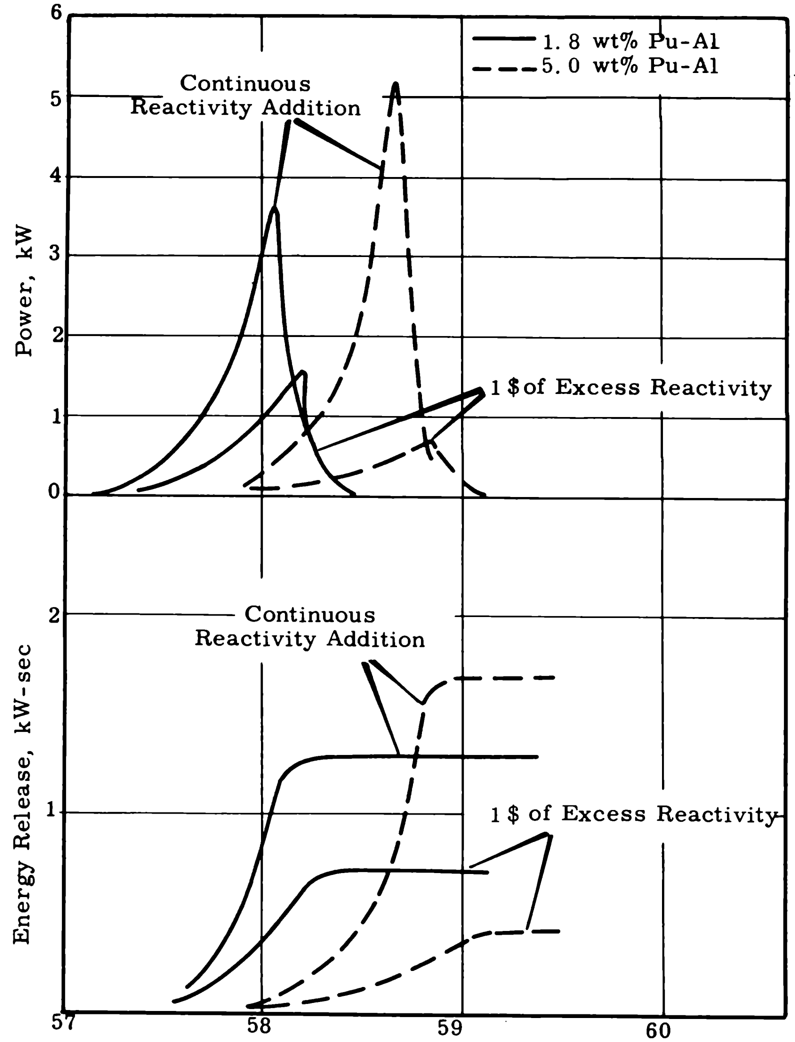

FIGURE 4. 3

Pu-Al Core Transient,
Safety Circuit Trip at 150 W

(2) Safety Circuit Fails to Trip (Reactor Initially 10 mk Sub-Critical

Two different types of excursions were studied. In one case the excess reactivity was limited to 1 $. The remaining case consisted of studies of continuous reactivity addition. As previously indicated, the latter case was studied to illustrate the consequences of excursions where the excess reactivity is greater than 1 $. The power level transient, energy release, and fuel element temperatures* for these excursions in a ceramic core loading are shown in Figure 4. 4. The power rise was overridden

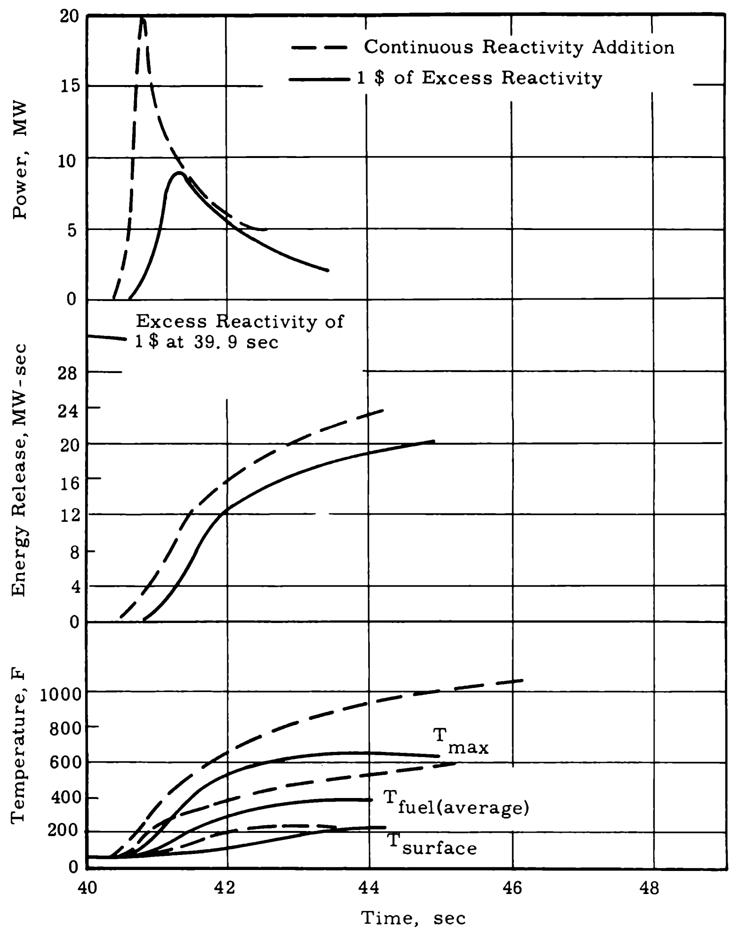

FIGURE 4. 4

PuO_2-UO_2 Core Transient, Safety Circuit Fails to Trip

*The maximum fuel element temperaure was calculated assuming that the maximum heat-generation rate (power density point) was twice that of the average heat-generation rate.

solely by the Doppler effect for both of the cases studied (continuous reactivity addition and excess reactivity limited to 1 \$). The effect of void formation does not contribute to reactivity feedback until the fuel element surface temperature is greater than 212 F and, as shown in Figure 4.4, the surface temperature does not reach 212 F until after the power peak. For the case of continuous reactivity addition, 3.6 mk of excess reactivity was added before the Doppler effect began to reduce the reactivity. The entire ramp, 3.34 mk, was added in the case where total excess reactivity was limited to 1 \$. The negative reactivity required to override the transient was 15 to 20% of the excess reactivity added.

The results for the Pu-Al core loadings are shown in Figures 4.5 and 4.6. All of the transients were overridden almost immediately upon the onset of boiling except in the case of continuous reactivity addition in the 5 wt% plutonium core. The void volume required to override the transients for the 1.8 wt% case was approximately 2% of the total moderator volume (0.1 ft^3). This volume corresponds to a negative reactivity of approximately 25% of the excess reactivity added ($\sim$ 2.6 mk of excess reactivity for the case of continuous addition and 2.1 mk of reactivity when the ramp was limited to 1 \$ of excess reactivity). The void volume required to override the transient for the 5 wt% plutonium case with \$1 of excess reactivity was approximately 1% of the total moderator volume (0.03 ft^3). This volume corresponds to a reactivity reduction of approximately 12%. Approximately 0.06 ft^3 of steam, 2% of the total moderator volume, was needed to override the transient for the 5 wt% plutonium case with continuous reactivity addition (corresponds to approximately 23% reduction of the 2.6 mk added). Larger void volumes are required to override transients for the 1.8 wt% plutonium case than for the 5.0 wt% plutonium case because of the larger moderator volume for the 1.8 wt% case (the same void coefficient was used for both cases).

The results of transient studies where the moderator void coefficient was arbitrarily reduced from the calculated value are shown in Figure 4.7 for the Pu-Al core containing 1.8 wt% plutonium. The excess reactivity

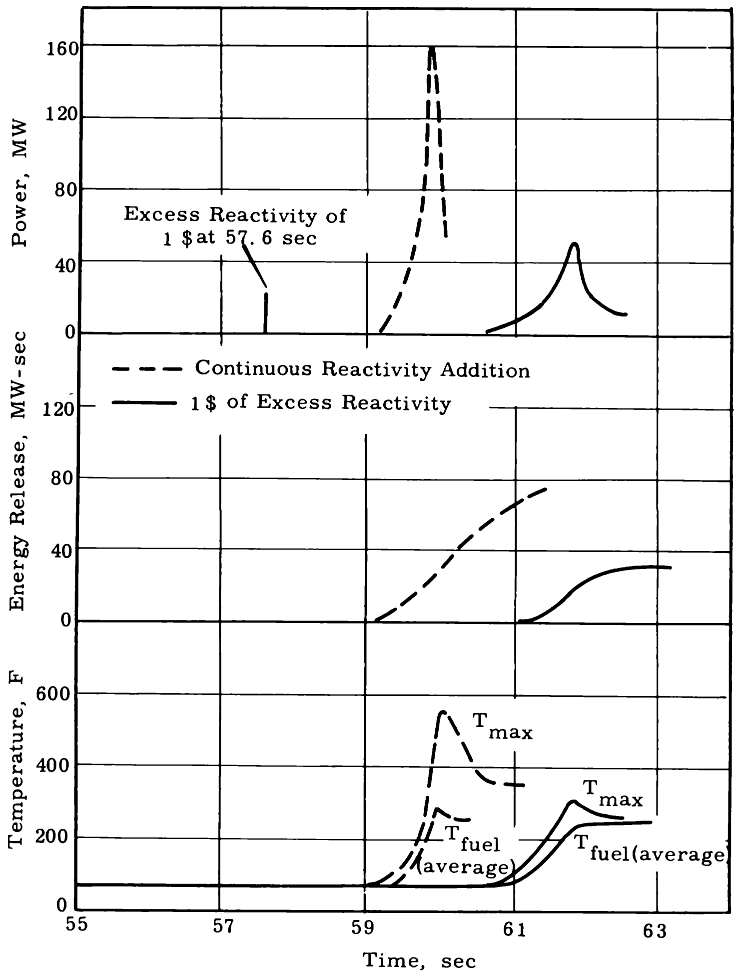

FIGURE 4.5

1.8 wt% Pu-Al Core Transient, Safety Circuit Fails to Trip

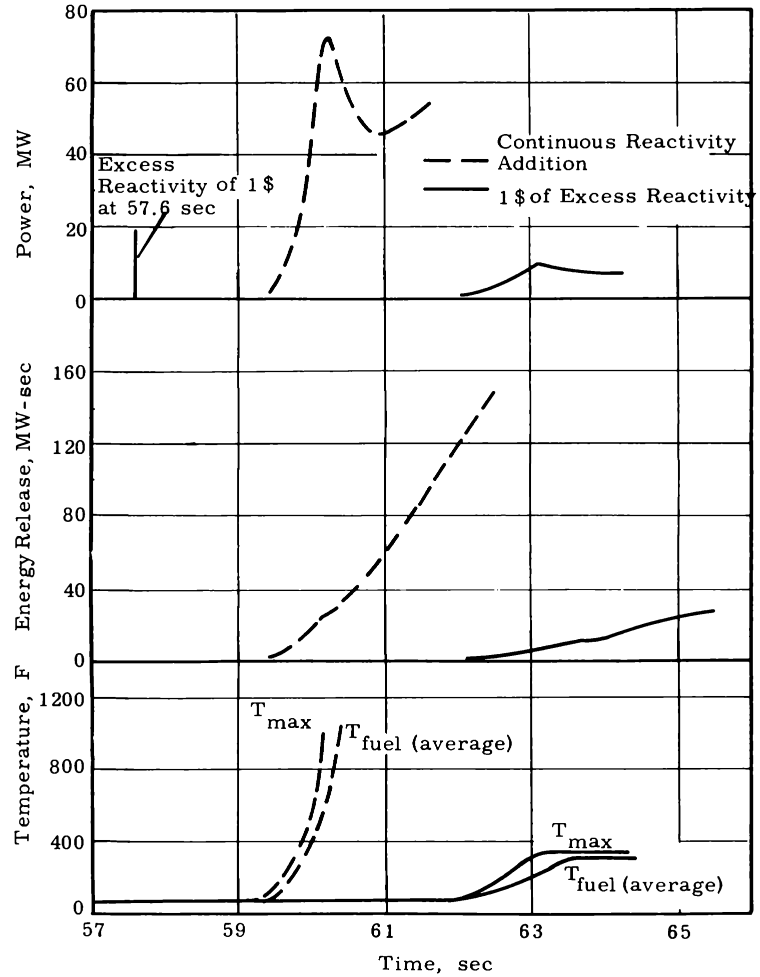

FIGURE 4.6

5 wt% Pu-Al Core Transient, Safety Circuit Fails to Trip

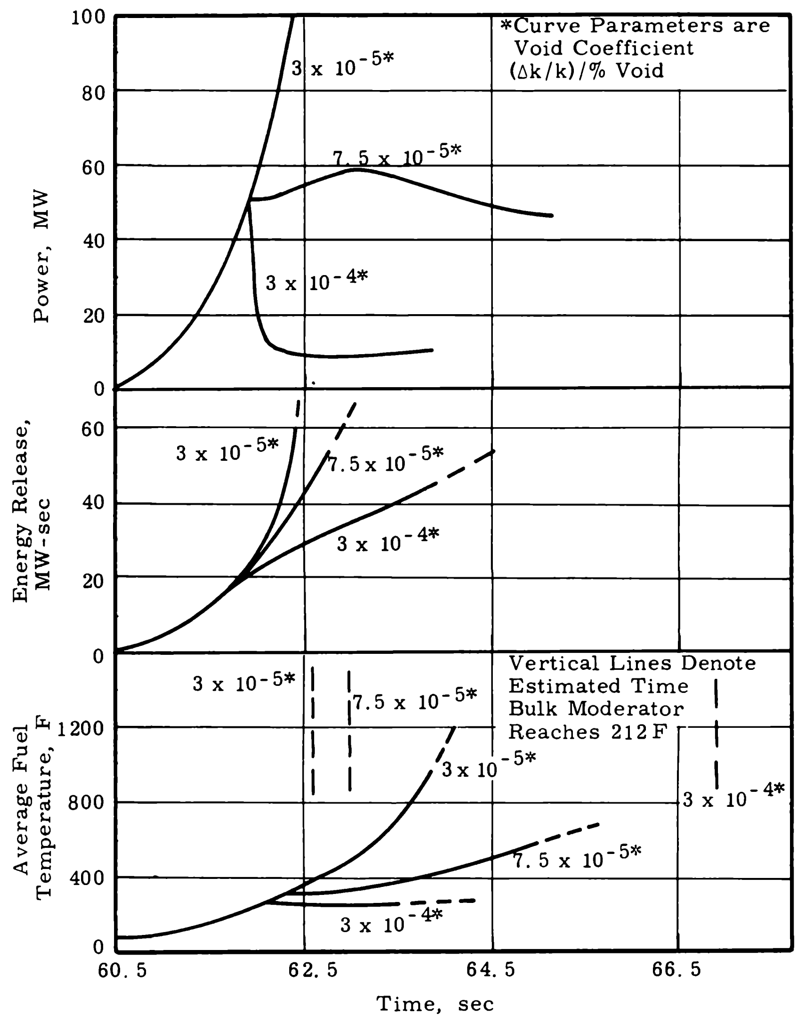

FIGURE 4. 7

Effect of Reducing Void Coefficient on Metallic Core Transient

was limited to \$1. As shown in Figure 4. 7, a reduction by a factor of 10^{-3} x 10^{-5} (Δk/k)/(% void) will provide an inherent negative reactivity effect too weak to override the transient. However, it is of interest to note that the bulk moderator temperature reached the saturation temperature before the maximum fuel temperature reached the melting point of the Pu-Al cores for both cases where the void coefficient was reduced. No attempt to estimate the reactivity effect resulting from bulk boiling of the moderator was included in the analog simulation.

The transient studies for continuous reactivity addition at a rate of 10 ¢ /sec indicate that the temperature dependent negative reactivity feedback effects limit the net positive reactivity at values only slightly greater than 1 \$ (approximately 1.1 \$ for the ceramic core and 1.2 to 1.3 \$ for the Pu-Al cores).

b. Operating Accidents

(1) Safety Circuit Trip at 150 W

The excursions terminated by scram and initiated while the reactor is operating ($k_{eff} = 1$) at 100 W, the maximum planned operating power for light water moderated experiments, would be less severe than those excursions initiated when the reactor is subcritical. This is because the reactor power is initially closer to the trip point and there is less time for reactivity addition. For the subcritical studies, criticality would be achieved at a flux level less than a level equivalent to 100 W and the reactor would already be on a positive period at the time a flux level equivalent to 100 W is reached.

(2) Safety Circuit Fails to Trip

For an excess reactivity limit of 1 \$, the severity of excursions initiated with the reactor subcritical and excursions initiated with the reactor operating at 100 W ($k_{eff} = 1$) would be almost the same. This is because the reactivity addition rate and response characteristics of the reactor to the reactivity disturbance, whether it is initially operating or subcritical, are such that the entire ramp of reactivity addition can be added before temperature dependent inherent reactivity effects begin to reduce the excess reactivity.

The excursions initiated from 100 W ($k_{eff} = 1$) would be less severe than excursions initiated from subcritical for continuous reactivity addition because more reactivity can be added to the subcritical case before temperature dependent reactivity effects override the excursion. For the subcritical case, criticality would be achieved at a flux level less than a level equivalent to 100 W and the reactor would already be on a positive period at the time a flux level equivalent to 100 W is reached.

c. Other Core Loadings

The experimental program will include tests of ceramic core loadings with enrichment levels other than that used for the transient study presented in this report. In general, for the PRCF reactivity addition and excess reactivity limits, the magnitude of the power peak would be slightly higher for all ceramic cores with a lower plutonium enrichment. The increased power at the time the excursion is overridden would be primarily the result of a decreased rate of fuel temperature rise because of increased core volume.

In core loadings using ceramic fuel of relatively low enrichment, Pu-Al fuel elements will be used as spike enrichment to provide sufficient reactivity to achieve criticality. The results of the studies presented in this report can be used to estimate the consequences of excursions in core loadings consisting of both ceramic fuel elements and spike fuel elements. As previously indicated, the excursions in all ceramic cores were overridden by the Doppler effect and the excursions in the all metallic cores were overridden by the void effect. The Doppler coefficient would be reduced for core loadings containing a large fraction of Pu-Al fuel elements, but the exponential power rise of an excursion for this type of loading would still be either overridden by the Doppler effect alone or initially slowed by the Doppler effect and then overridden at the onset of boiling at the surface of the metallic fuel. It should be noted that the value of the thermal time constant for the metallic fuel (0.06 sec) is so small that the lag of the surface temperature behind the core temperature is insignificant for the reactor periods studied. The effect of the physics and thermal constants characteristic of a mixed ceramic-metallic fuel loading is illustrated by the following example. It was assumed that the core contained approximately equal numbers of ceramic and metallic fuel elements. Assuming that the ceramic fuel was 0.75 wt% PuO_2 in natural UO_2 it was calculated that approximately two-thirds of the reactor power was generated in the ceramic fuel. It can be shown that the Doppler effect accompanying a fuel temperature change from T_o to T can be expressed as

$$\Delta k/k = C_f (T^{\frac{1}{2}} - T_o^{\frac{1}{2}})$$

where T is in degrees K.[1] Calculations indicate that the coefficient, C_f, for the mixed loading would be approximately one-half that of a core loading consisting entirely of ceramic fuel. The transient studies presented in this report indicate that approximately 0. 6 mk of negative reactivity is required to override excursions in cores with similar physics constants. For the limited excess reactivity permitted in the PRCF, changes in the magnitude of the total delayed neutron fraction and the prompt neutron lifetime would have little effect on the character of the transient. The coefficient, C_f, used for the studies of all ceramic cores presented in this report was approximately -0.6×10^{-3}. An additional temperature increase of approximately 80 F* would be required to compensate for a 50% reduction in the coefficient to to achieve a negative Doppler effect of 0. 6 mk. Since the "thermal capacitance," the product of the fuel mass and fuel heat capacity, of both types of fuel is approximately the same value, and the heat generation rate of the ceramic fuel is twice that of the Pu-Al fuel, the rate of fuel temperature rise for the metallic fuel would be about one-half that of the ceramic fuel. It would be expected that a transient in this postulated core would be overridden by the Doppler effect before the metallic fuel surface temperature reached 212 F.

d. Discussion

The results, peak power and energy release, of studies presented in this report are compared to those previously presented in Supplements II and III in Table 4. 2. The results given in Table 4. 2 are for cases where the excursion was initiated from subcritical and the excess reactivity was limited to 1 $.

As shown in Table 4. 2, the peak excursion power for the ceramic core loading (1. 5 wt% PuO_2 in UO_2, natural uranium) was 9 MW for the case where it was assumed that the safety circuit failed. This power is of the same order of magnitude as was obtained for a similar excursion in a typical PuO_2-UO_2, EBWR core loading (1. 5 wt% PuO_2 in depleted UO_2,

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(1) R. E. Peterson. Nuclear Parameters - PRTR Mixed Oxide Fuel, Safeguards Analysis, HW-74346. July 18, 1962.

* For C_f of 0.26×10^{-3}, $\Delta k/k$ is 0.6×10^{-3} where T is 378 K (222 F) and T_o is 294 K (70 F).

0.22 at.% U^{235}). The different physics and thermal constants of the two PuO_2-UO_2 core loadings have little effect on the magnitude of the initial power peak.

TABLE 4.2

COMPARISON WITH PREVIOUS STUDIES

Core Loading	Safety Circuit Trips: Peak Power, kW	Safety Circuit Trips: Energy Release, kW-sec	Safety Circuit Does Not Trip: Power at Time Transient Overridden, MW	Safety Circuit Does Not Trip: Energy Release at Time of Initial Power Peak, MW-sec
2.5 wt% PuO_2 and Depleted UO_2 [a, b]	83	18	25	6
2.5 wt% PuO_2 and Depleted UO_2 [a, c]	25	4.5	5	4.5
1.5 wt% PuO_2 and Depleted UO_2 [a, d]	6	2.1	---	---
1.5 wt% PuO_2 and Depleted UO_2 [a, e]	3	1.1	4	2
1.5 wt% PuO_2 and Natural UO_2 [f]	8	2.8	9	4
Pu-Al, 1.8 wt% Pu [g]	1.6	0.7	50	15
Pu-Al, 5.0 wt% Pu [g]	0.7	0.4	10	5

(a) Studies of EBWR core loadings.
(b) See Supplement II, Figure 2.9 and Figure 2.11, lattice spacing of 0.71 in.
(c) See Supplement II, Figure 2.9 and Figure 2.11, lattice spacing of 0.81 in.
(d) See Supplement III, Figure 3.1, lattice spacing of 0.71 in.
(e) See Supplement III, Figure 3.1 and Figure 3.3, lattice spacing of 0.81 in.
(f) See Figures 4.2 and 4.4.
(g) See Figures 4.3, 4.5, and 4.6.

The results of the transient studies, as shown in Table 4.2, indicate that, in general, the magnitude of the initial power peak will be of the same order of magnitude for core loadings with relatively strong inherent negative reactivity effects, for the case where it was assumed that the safety circuit fails. This is primarily the result of the excess reactivity limit, the reactivity addition rate limit, and the thermal properties of the metallic fuel. The magnitude of the initial power peak is relatively independent of the enrichment level of the fuel for core loadings with strong inherent negative reactivity effects. It should also be noted that because of the excess reactivity limit of 1 \$, results obtained for cases where it is assumed that the reactivity is added at a faster rate, again assuming safety circuit failure, would be approximately the same, even if it is assumed that 1 \$ of excess reactivity is added as a step. This is because the response characteristics of core loadings studied allow the complete insertion of 1 $ of excess reactivity at a rate of 10 ¢/sec before temperature dependent inherent reactivity effects begin to reduce the excess reactivity.

It is not meant to imply by the extensive analysis of cases where it is assumed that the safety circuit fails (as opposed to the analysis of excursions terminated by the safety circuit) that it is believed that complete failure of the safety circuit is likely to occur.

3. <u>Effect on Maximum Credible Accident</u>

The analyses presented in this document indicate that operation of the facility in accordance with the proposed operating limits of Part C with either ceramic fuel elements, (Pu-Al) fuel elements, or mixtures of the two would not lead to a nuclear accident in which fuel melting would occur. The severity of the credible accidents for light water moderated experiments should be less than that of the maximum credible accident described in HW-69168 and Supplement I.

APPENDIXES

APPENDIX A

LETTER: RICHLAND OPERATIONS OFFICE TO GENERAL ELECTRIC COMPANY

United States
Atomic Energy Commission
Hanford Operations Office
P. O. Box 550
Richland, Washington

Jun 6 1962

RF:HAH

General Electric Company
Hanford Atomic Products Operation
Richland, Washington

Attention: H. M. Parker, Manager
Hanford Laboratories Operation

Subject: PLUTONIUM RECYCLE CRITICAL FACILITY FINAL SAFEGUARDS ANALYSIS - HW-69168

Gentlemen:

Our review of the subject analysis indicates a need for additional supporting data, backed up with calculating methods, appropriate physics constants and assumptions, including hazards analyses of representative fuel loadings given on page 8 of Document HW-69168. Minimum information provided should be similar to that shown on pages 57, 58, 68, 69, 100, and 101, and should include pertinent physics constants as follows:

1. Delayed neutron fraction.
2. Temperature coefficients.
3. Void coefficient.

Assuming that the safety circuit may not function, please present a complete analysis of the radiological consequences of an excursion with a fully enriched plutonium reactor core loading or a more hazardous reactor core loading, if any.

Information is needed explaining in some detail your plans for minimizing operational mishaps, investigating unusual or unexpected incidents, and recovery techniques used in the event of a maximum credible accident. To help clarify relative locations of alarm devices with respect to the operating console, and the aqueous effluent and ventilation inter-ties between the Plutonium Recycle Critical Facility and the Plutonium Recycle Test Reactor, please furnish us simplified one-line diagrams showing these relationships. Operational mishaps should include analysis of fire from nuclear or non-nuclear incidents and the availability of fire control equipment. Please furnish drawings and other supporting data on 8-1/2" x 11" size paper.

It is stated in the Safeguards document that the Zircaloy fuel cladding will contain molten fuel elements (Pu-Al) up to 48 minutes and that a short-cooled element hanging in air will reach up to 900 °C. What effect would the corrosivity of molten aluminum, fission gas release, and fuel element exposure have on the integrity of the Zircaloy fuel cladding?

Since one of the weakest points in the PRCF design is the flexible cooling hoses for the short-cooled fuel element, please advise what proof tests were made on these hoses to assure integrity.

Very truly yours,

/s/ P. G. Holsted

Paul G. Holsted, Director
Civilian Reactor Development
and Research Division

APPENDIX B

ANALYTICAL FORMULATION OF NEUTRON KINETICS

Table of Nomenclature

n = neutron density, neutrons/cm^3
t = time, sec
β = total delayed neutron fraction
l^* = mean effective neutron lifetime, sec
λ_i = precursor delay constant of i^{th} kind, sec^{-1}
C_i = precursor nuclei/cm^3 (i^{th} kind)
β_i = fraction of delayed neutrons of i^{th} kind
P = average power level, W
V = volume of fuel, ft^3/fuel element
N = number of fuel elements
p = resonance escape probability at T_o
T_o = initial fuel temperature, F
T_1, T_2 = average temperature of fuel, F
M = weight of fuel, lb/fuel element
C_p = heat capacity of fuel, W-sec/(F)(lb)
h = heat transfer coefficient, W/(ft^2)(F)
A = heat transfer area of fuel, ft^2/fuel element
T_{SAT} = saturation temperature of moderator, F
V_S = volume of vapor formed in coolant zone by heat transfer, ft^3
V_R = total volume of gas formed by radiolytic decomposition, ft^3
V_{CR} = volume of gas formed in coolant by radiolytic decomposition, ft^3
V_C = total volume of coolant zone, ft^3/fuel element
V_{MR} = volume of gas formed in moderator by radiolytic decomposition, ft^3
V_M = volume of moderator zone, ft^3/fuel element
$\delta p/\delta x$ = pressure gradient associated with expulsion, lb/ft^2
ρ_L = density of moderator, lb/ft^3
ρ_V = density of D_2O vapor, lb/ft^3
L = active length of fuel core, ft
g = gravitational constant, ft/sec^2

v = kinematic viscosity, D_2O vapor, ft^2/sec
k = thermal conductivity, D_2O vapor, Btu/(ft)(sec)(F)
λ = latent heat of vaporization, Btu/lb
Subscript u = denotes UO_2 fuel
Subscript p = denotes Pu-Al fuel
Subscript z = denotes zirconium cladding
C_1 = fraction of fissions occurring in U^{235}
C_2 = fraction of fissions occurring in Pu^{239}
C_3 = coolant void coefficient, $\Delta k/k$ for 100% loss
C_4 = moderator void coefficient, $\Delta k/k$ for 100% loss

The neutron density as a function of time is given by:

$$dn/dt = \frac{\Delta k - \beta}{l^*} n + \sum_{i=1}^{7} \lambda_i C_i + S$$

$$\Delta k = k_{eff} - 1$$

$$\frac{dC_i}{dt} = \frac{\beta_i}{l^*} n - \lambda_i C_i$$

$$\beta_i = C_1 \beta_i(U^{235}) + C_2 \beta_i(Pu^{239})$$

$$\beta = \sum_{i=1}^{7} \beta_i$$

Average fuel element temperatures as a function of time are given by:

$$dT_1/dt = \frac{C_1 P}{(M_z C_{pz} + M_u C_{pu}) N_u} - \frac{h N_u A (T_1 - T_{SAT})}{(M_z C_{pz} + M_u C_{pu}) N_u}$$

$$dT_2/dt = \frac{P}{\left[\left(\frac{N_uM_uC_{pu} + N_pM_pC_{pp}}{N_u + N_P}\right) + M_zC_{pz}\right]N}$$

$$- \frac{hNA\ (T - T_{SAT})}{\left[\left(\frac{N_uM_uC_{pu} + N_pM_pC_{pp}}{N_u + N_p}\right) + M_zC_{pz}\right]N}$$

Negative reactivity effects were obtained from the following equations:

$$\Delta k\ (\text{Doppler}) = -\ (0.74 \times 10^{-2}) \ln \frac{(P_uN_u + P_pN_p)}{N_u + N_p} \left[(0.555\, T_1 + 255)^{\frac{1}{2}} - (0.555\, T_o + 255)^{\frac{1}{2}}\right]$$

$$\Delta k\ (\text{Coolant Void}) = -C_3 \frac{(V_S + V_{CR})}{NV_C}$$

$$\Delta k\ (\text{Moderator Void}) = -C_4 \frac{(V_{MR})}{NV_M}$$

where the void volume formed is given by:

$$V_S = 4/5\, N_pA\left(\frac{\delta p}{\delta x} + \rho_L - \rho_V\right)^{\frac{1}{4}} \left[\left(\frac{4L}{g}\right)\left(\frac{vk}{\lambda}\right)(T_2 - T_{SAT})\right]^{\frac{1}{4}}$$

$$V_R = 2.9 \times 10^3 \int \Delta P\, dt;\ V_{CR} = 0.05\, V_R;\ V_{MR} = 0.95\, V_R\ .$$

Delayed Neutron Data

Group	U^{235} λ_i (a)	U^{235} β_i	Pu^{239} λ_i (b)	Pu^{239} β_i
1	0.0124	0.02105×10^{-2}	0.0128	0.007216×10^{-2}
2	0.0305	0.14008×10^{-2}	0.0301	0.06254×10^{-2}
3	0.1110	0.12551×10^{-2}	0.124	0.04433×10^{-2}
4	0.3010	0.25263×10^{-2}	0.325	0.06838×10^{-2}
5	1.13	0.07368×10^{-2}	1.12	0.01787×10^{-2}
6	3.00	0.02672×10^{-2}	2.69	0.00928×10^{-2}
7 (photoneutrons)	0.0195	0.0493×10^{-2}	0.0195	0.04180×10^{-2}
		0.68897×10^{-2}		0.2514×10^{-2}

(a) Delay constants for U^{235} were used for studies with 40% and 65% of fissions in U^{235}.

(b) Delay constants for Pu^{239} were used for studies with 80% of fissions in Pu^{239}.

APPENDIX C

SUMMARY OF THE METHOD OF ANALYSIS FOR ACCIDENT STUDIES

Table of Nomenclature

A	=	Heat transfer area of the fuel
C_1	=	Reactor void coefficient
C_2	=	Reactor fuel temperature coefficient
C_i	=	Concentration of delayed neutron precursors of i^{th} kind
C_{pF}	=	Mean heat capacity of fuel elements and jackets (weight of fuel times the heat capacity of fuel plus weight of jackets times the heat capacity of the jackets)
C_{pm}	=	Mean heat capacity of moderator
g	=	Acceleration due to gravity
h	=	Boiling heat transfer coefficient
k	=	Neutron multiplication factor
k_f	=	Thermal conductivity of fuel
k_s	=	Thermal conductivity of steam
K	=	Proportionality constant
l^*	=	Neutron lifetime
L	=	Length of active section of core (fuel)
n	=	Neutron density
p	=	Pressure
P	=	Reactor power
S	=	Source strength
t	=	Time
T	=	Reactor period
T_c	=	Fuel element core temperature, average
T_i	=	Initial temperature
T_m	=	Moderator temperature, average
T_s	=	Fuel element surface temperature, average
T_{sat}	=	Moderator saturation temperature
v	=	Neutron velocity
V_f	=	Volume of fuel

V_m = Volume of moderator over which void formation is effective and upon which C_1 is based
V_r = Volume of void formed by radiolytic decomposition
V_s = Volume of steam formed by boiling
β = Total fraction of neutrons that are delayed
β_i = Fraction of delayed fission neutrons of i^{th} kind
ρ_m = Moderator density
ρ_s = Steam density
λ = Heat of vaporization
λ_i = Precursor delay constant of i^{th} kind
ν_s = Kinematic viscosity of steam
Δk = Excess reactivity
Δk_{ft} = Doppler reactivity effect
Δk_{in} = Reactivity disturbance
Δk_{mv} = Moderator void reactivity effect
Δk_{mt} = Moderator density reactivity effect
Σ_f = Macroscopic fission cross section
$\delta p/\delta x$ = Pressure gradient associated with expulsion of moderator

The neutron kinetic equations presented later in this appendix were used to study accidents which were overridden by rod insertion and inherent shutdown mechanisms. The calculated negative reactivity insertion upon rod drop is shown in Figure 2. 7.

For the studies wherein the excursions were overridden by inherent shutdown mechanisms (safety circuit failure), it was anticipated that the Doppler coefficient would override the excursion. It was assumed that the transients would be rapid enough so that heat transfer to the moderator could be neglected, until the fuel temperature reached the saturation temperature of the moderator. The instantaneous power generated in the moderator was also neglected.

Equations were included in the analysis to estimate the effect of the enhanced resonance absorption due to Doppler broadening in U^{238} and Pu^{240} and the effect of decreasing the moderator density by surface boiling, heat transfer to the bulk moderator, and radiolytic decomposition.

Other assumptions used in the safety circuit failure studies included:

- Equations used to determine the amount of void volume formed by surface boiling were formulated from a model for film boiling on vertical plane surfaces described by Janssen, et al.[1] The formulation was used to determine the void volume formed during both nucleate and film boiling conditions.
- The sensible heat and heat of vaporization in void formation were neglected.
- The thermal properties of the core components remain constant throughout the transient.
- The fuel element surface boiling heat transfer coefficient used was adapted from data presented in Bonilla[2] for quenching cylinders (L/D = 1) in cold water and is shown in Figure C-1.

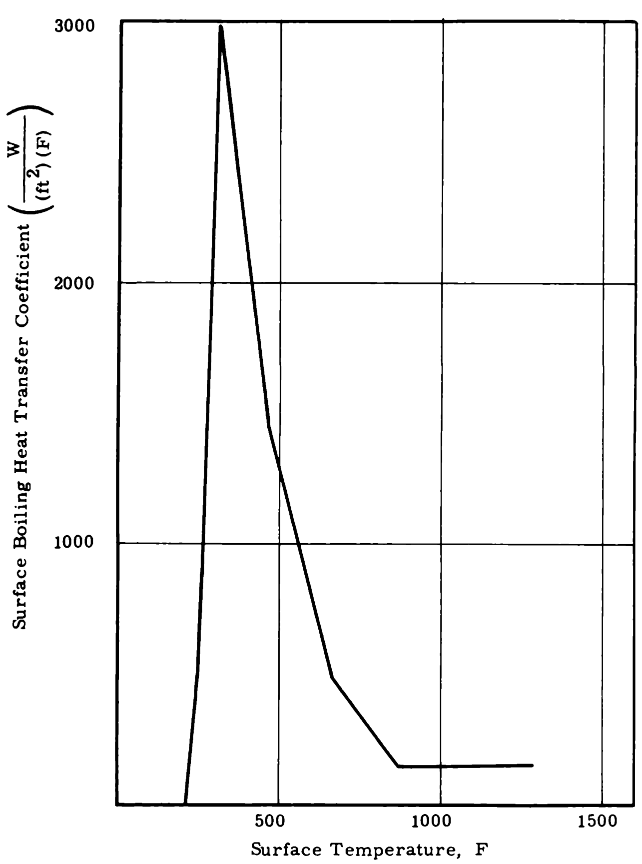

FIGURE C-1
Boiling Heat Transfer Coefficient

(1) E. Janssen, W. H. Cook, and K. Hikodo. Metal Water Reactions: I, A Method for Analyzing a Nuclear Excursion in a Water Cooled and Moderated Reactor, GEAP-3073. October 15, 1958.

(2) C. F. Bonilla. Nuclear Engineering, McGraw Hill Book Co., Inc. New York, N.Y. 1957.

- The surface temperature of the fuel element was calculated from an equation derived from the steady state equation for radial heat transmission in a cylinder with internal heat generation, i. e., at any time t, the fraction of power lost from the fuel was proportional to $hA\Delta T$. Since heat transfer to the moderator was neglected (h = o) until the saturation temperature was reached, the fuel surface and core temperature were identical until the surface temperature reached 212 F.

The equations used to simulate the accident studies are summarized below.

Neutron Kinetics

$$dn/dt = (\Delta k - \beta)\frac{n}{1^*} + \sum_{i=1}^{n} \lambda_i C_i + S \tag{1}$$

where

$$\Delta k = \Delta k_{in} + \Delta k_{rods} + \Delta k_{mv} + \Delta k_{mt} + \Delta k_{ft}$$

and

$$\beta = \sum_{i=1}^{n} \beta_i$$

$$dC_i/dt = \beta_i \frac{n}{1^*} - \lambda_i C_i \tag{2}$$

$$P = K\Sigma_f nvV_f. \tag{3}$$

Temperature Equations

$$dT_c/dt = \frac{P}{C_{pF}} - hA\frac{(T_s - T_i)}{C_{pF}} \tag{4}$$

$$T_s = T_c - \left(hA\ \frac{(T_s - T_i)}{L}\right)\left(\frac{1}{8\pi k_f}\right) \tag{5}$$

$$dT_m/dt = hA\frac{(T_s - T_i)}{C_{pm}} \tag{6}$$

Inherent Shutdown Mechanisms

$$V_s = \frac{4A}{5}\left(\frac{\delta p}{\delta x} + \rho_m + \rho_s\right)^{-\frac{1}{4}} \left[\left(\frac{4L}{g}\right)\left(\frac{\nu_s k_s}{\lambda}\right)\left(T_s - T_{sat}\right)\right]^{\frac{1}{4}} \quad (7)$$

$$V_r = \frac{K}{P}\int P dt \quad (8)$$

$$\Delta k_{mv} = C_1 \; 100 \left(\frac{V_s + V_r}{V_m}\right) \quad (9)$$

$$\Delta k_{mt} = C_1 \; 100 \left[\frac{\rho_{m'}(T_i) - \rho_{m'}(T_m)}{\rho_{m'}(T_i)}\right] \quad (10)$$

$$= C_1 \; (0.02244 \; T_m - 1.571)$$

where T_m is in degrees F and $C_1 = \Delta k/k/\%$ void

$$\Delta k_{ft} = C_2\left(\sqrt{T_c} - \sqrt{T_i}\right) \quad (11)$$

APPENDIX D

HEAT TRANSFER EQUATIONS

The similarity in response for transients overridden by inherent shutdown mechanisms, especially after the power peak occurs, is the result of using a "resistance-capacitance" model to derive the heat transfer equations. For the study presented in this report, separate equations were used to describe the rate of temperature change of the fuel element core and fuel element cladding. This was done to better define the lag existing between the two temperatures during the transient. The use of the "resistance-capacitance" model allows the fuel element core to achieve higher temperatures before the fuel element surface temperature reaches 212 F (and steam formation becomes a shutdown mechanism). Since fuel element core temperatures are higher, the Doppler effect is stronger, even after the power peak. The results of the transient studies for the 1.5% case indicate that the excess reactivity is almost reduced to zero before the fuel element surface temperature reaches 212 F. The Doppler effect is the primary factor in both overriding the power transient and determining the response of the transient immediately after the power peak.

The thermal capacitances (heat capacity effect) of the fuel element core and cladding were "lumped" and the effect of core and clad thermal conductivity and the core-clad heat transfer coefficient on thermal resistance were neglected in the equations presented in Supplement II. In addition, equations were such that fuel element surface and core temperature were identical until the surface temperature reached 212 F. Since heat transfer by local boiling at the fuel element surface becomes significant at temperatures slightly greater than 212 F, the temperature rise and Doppler effect was essentially limited at this point. The Doppler effect was the mechanism for initially overriding the power rise for the transients presented in Supplement II, but sufficient excess reactivity remained after the fuel element core temperature reached equilibrium to reinitiate the transient. Any remaining excess reactivity was essentially reduced to zero (and the transient overridden again) upon the onset of surface boiling for cores with a negative moderator void coefficient. The power level rise continued for cores with a zero void coefficient.

Formulation of Equations

Table of Nomenclature

A_c = Cross sectional area of cladding (per rod)

A_f = Cross sectional area of fuel (per rod)

A_m = Cross sectional area of moderator (associated with a single fuel rod)

A_s = Surface area of fuel rod

C_1 = Moderator reactivity coefficient (void and temperature)

C_2 = Fuel (Doppler) temperature reactivity coefficient

C_i = Concentration of delayed neutron precursor of i^{th} kind

C_{pc} = Heat capacity of fuel element cladding

C_{pf} = Heat capacity of fuel element core

C_{pm} = Heat capacity of moderator

g = Gravitational constant

h_1 = Fuel-clad heat transfer coefficient

h_2 = Fuel element surface heat transfer coefficient

k_f = Thermal conductivity of fuel

k_s = Thermal conductivity of steam

K = Proportionality constant

ℓ = Prompt neutron lifetime

L = Length of active section of core (fuel)

n = Neutron density

N = Total number of fuel rods per core

P = Average reactor power

q = Nuclear heating (per foot of fuel rod)

r_c = Radius of clad fuel

r_f = Radius of fuel

r_{cell} = Radius of moderator associated with a lattice position, upon which C_1 is based

S = Source strength

t = Time

t_c = Fuel clad thickness

T_f = Average fuel temperature
T_i = Initial temperature
T_m = Bulk moderator temperature
T_s = Average clad (surface) temperature
T_t = Temperature at time t
T_b = Saturation temperature of moderator
V = Volume of moderator upon which C_1 is based
V_s = Void volume formed by boiling
β = Total fraction of neutrons that are delayed
β_i = Fraction of delayed neutrons of i^{th} kind
ρ_c = Density of fuel element cladding
ρ_f = Density of fuel element core
ρ_m = Density of moderator (liquid)
ρ_s = Density of steam
λ = Latent heat of vaporization of moderator
λ_i = Precursor delay constant (of i^{th} kind)
υ_s = Kinematic viscosity of steam
$\Delta k/k$ = Excess reactivity
$\Delta k/k_{fuel}$ = Fuel temperature (Doppler) reactivity effect
$\Delta k/k_{in}$ = Reactivity disturbance
$\Delta k/k_{mod}$ = Moderator temperature reactivity effect
$\Delta k/k_{scram}$ = Reactivity effect of safety system
$\Delta k/k_{void}$ = Reactivity effect of steam (void) formation

Assumptions used in the formulation of equations are the same as those presented in the appendix of Supplement II except for the use of different heat transer (rate of temperature change) equations. Additional assumptions used in the formulation of the heat transfer equations included:

- Axial heat transfer is neglected.
- The thermal resistance of the cladding is determined by the fuel element surface heat transfer coefficient. The thermal conductivity of the clad is neglected.
- Heat transfer from the reactor tank to the cell is negligible during the transient.

The equations used to simulate the excursions are summarized below:

Neutron Kinetics

$$dn/dt = (\Delta k/k - \beta)(n/\ell) + \sum_{i=1}^{n} \lambda_i C_i + S \tag{1}$$

where

$$\Delta k/k = \Delta k/k_{in} + \Delta k/k_{scram} + \Delta k/k_{void} + \Delta k/k_{fuel} + \Delta k/k_{mod} \tag{2}$$

and

$$\beta = \sum_{i=1}^{n} \beta_i \tag{3}$$

$$dC_i/dt = \beta_i/\ell - \lambda_i C_i \tag{4}$$

Rate of Temperature Change

$$\frac{dT_f}{dt} = \frac{q}{A_f C_{pf} \rho_f} - \frac{T_f - T_s}{A_f C_{pf} \rho_f \left[\frac{1}{8\pi k_f} + \frac{1}{2\pi r_f h_1}\right]} \tag{5}*$$

where

$$q = P/NL = Kn/NL \tag{6}$$

and

$$A_f = \pi r_f^2 \tag{7}$$

$$\frac{dT_s}{dt} = \frac{T_f - T_s}{A_c C_{pc} \rho_c \left[\frac{1}{8\pi k_f} + \frac{1}{2\pi r_f h_1}\right]} - \frac{T_s - T_m}{A_c C_{pc} \rho_c \left[\frac{1}{2\pi r_c h_c}\right]} \tag{8}*$$

where

$$A_c \simeq 2\pi r_c t_c \tag{9}$$

---

*The "maximum" fuel element core temperature is obtained from equations similar to (5) and (8) except that a value for the flux peaking factor is inserted into the numerator of the first term on the right side of Equation (5). T_f and T_s are then defined as the maximum fuel element core and surface temperature.

$$\frac{dT_m}{dt} = \frac{T_s - T_m}{A_m C_{pm} \rho_m \left[\frac{1}{2\pi r_c h_2}\right]} \tag{10}$$

where

$$A_m = \pi r_{cell}^2 - \pi r_c^2 \tag{11}$$

Inherent Reactivity Effects

$$V_s = 0.8\, N A_s (\rho_m - \rho_s)^{-\frac{1}{4}} \left[\left(\frac{4L}{g}\right)\left(\frac{\upsilon_s k_s}{\lambda}\right)(T_s - T_b)\right]^{\frac{1}{4}} \tag{12}$$

where

$$A_s = 2\pi r_c L \tag{13}$$

$$\Delta k/k_{void} = \frac{100\, C_1 V_s}{N_V} \tag{14}$$

where

$$V = \pi L (r_{cell}^2 - r_c^2) \tag{15}$$

$$\Delta k/k_{fuel} = C_2 (T_{f,t}^{\frac{1}{2}} - T_{f,i}^{\frac{1}{2}}) \tag{16}$$

$$\Delta k/k_{mod} = C_1 (0.02244\, T_{m,t} - 1.571 \quad . \tag{17}$$

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