HAZARDS EVALUATION REPORT

FOR

TIGHTLY-PACKED FUEL STORAGE

CRITICAL EXPERIMENT PROGRAM

LICENSE NO. CX-10

By

G. S. Hoovler

October 23, 1979

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1.0 INTRODUCTION

Under a recent Department of Energy Contract (EY-77-C-09-1001 effective July 18, 1977 through July 17, 1979), B&W conducted critical experiments to benchmark criticality calculations of stored LWR nuclear fuel assemblies. Criticality measurements were made with low-enriched UO_2 pins arranged to simulate LWR assemblies in close proximity water storage. These measurements were made in the CX-10 facility located at the Company's Lynchburg Research Center.

This experimental effort has provided an excellent data base on which industry can rely for assuring safe close-packed storage of fuel assemblies. Since the program was conceived, however, the likelihood of eventual commercial nuclear fuel reprocessing in the U.S. has decreased. Tightly packed fuel pin storage has therefore become a more attractive option. Suitable experimental data do not exist to benchmark nuclear criticality codes under this high density fuel storage condition.

DOE has contracted B&W to broaden the current study to include tightly packed fuel pin storage, a concept that is applicable to both on site and away-from-reactor storage. This experimental work will also be performed at the Company's CX-10 facility. Experimental measurements will be made to provide benchmark criticality data for underwater storage of fuel pins packed one against the other in specially designed cannisters. This concept represents the most efficient way to utilize available storage space. If this mode of fuel storage were used, no movement of spent fuel from most reactor sites would be required over the entire reactor lifetime. In a recent study for the Department of Energy, Nuclear Assurance Corporation concluded that pin storage (if done at the reactor site) would also reduce the cost of shipping spent fuel to an away-from-reactor depository.

This program will require the use of highly under-moderated lattices (nonmoderator-to-moderator volume ratio = 1.4-9.7). Past programs at the CX-10 have studied lattices with M/W ratios no higher than 1.2. Accordingly, we have re-evaluated the maximum credible accident (MCA). We conclude the accident would not result in fuel pin failure or the release of radioactive effluent, even if a partial failure of the safety system is assumed.

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2. LICENSE HISTORY

2.1 Basic License

The original license was issued as CX-10 on December 13, 1957, and is contained in Docket 50-13. The license permitted experimental work with critical assemblies of slightly enriched UO_2 in stainless steel pins arranged in a light water moderated lattice (Reference 1).

2.2 Amendment #1

This amendment, issued May 7, 1959, authorized the use of segmented fuel pins containing slightly enriched UO_2 manufactured by General Electric (Reference 2).

2.3 Amendment #2

This amendment was issued June 25, 1959 and authorized the use of aluminum clad fuel pins containing a thoria-urania fuel mixture (Reference 3).

2.4 Amendment #3

Issued on February 7, 1961, this amendment authorized experiments conducted in a modified tank and water handling system that were to be used with the previously authorized fuels, but with mixtures of light and heavy water moderator to study the spectral shift concept (Reference 4).

2.5 Amendment #4

Issued on August 23, 1961, this amendment was informational and incorporated a description of how reactivity changes are made when the facility is in a shutdown condition (Reference 5).

2.6 Amendment #5

This amendment, issued May 31, 1962, authorized the use of aluminum clad fuel pins containing $2\frac{1}{2}\%$ enriched UO_2 (Reference 6).

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2.7 Amendment #6

This amendment, issued July 2, 1965, authorized the use of various poison and non-poison materials in the core, either in place of or between fuel pins (Reference 7).

2.8 Amendments 7 Through 10

Except for amendment 10, which authorized the possession and use of an AmBe neutron source for startup, these amendments were administrative. None had a significant impact on facility design or operating limits (References 8-11).

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3.0 SUMMARY

Three different fuel rod clusters will be assembled from 2.46% enriched, aluminum-clad UO_2 rods:

1. Cluster #1 will consist of a 15x17 matrix of fuel rods loaded on a triangular pitch equal to the fuel pin diameter (M/W = 9.7).

2. Cluster #2 will consist of a 15x15 matrix of fuel rods loaded on a square pitch equal to the fuel pin diameter (M/W = 3.7).

3. Cluster #3 will consist of a 13x13 matrix of fuel rods loaded on square pitch equal to 1.158 times the fuel pin diameter (M/W = 1.4).

A 5x5 array of each cluster type will be loaded in the 9 foot diameter core tank and brought critical. Water gaps of uniform width will be left between the clusters so that the array can be brought critical and to allow passage of safety blades.

Maximum power, rod worths, excess reactivity, shutdown margin, and reactivity addition rates will be consistent with current license limits. Systems and features that relate to safety (moderator system, control rods, control rod drives, safety instrumentation, etc.) will be the same as used in past programs.

Consideration of potential accidents established that the maximum credible accident would be the continuous addition of reactivity at the rate of .05%/sec. An analysis of this accident shows that the excursion would be terminated by scram without fuel rod rupture or the release of any radioactive effluent.

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4.0 DESCRIPTION OF CRITICAL EXPERIMENT PROGRAM

4.1 General

Critical loadings will consist of 25 fuel clusters grouped in a 5x5 array as shown in Figure 1. Within each fuel cluster, pins will be arranged in one of the following ways:

1. Triangular pitch equal to pin diameter
    (See Figure 2)
2. Square pitch equal to pin diameter
    (See Figure 3)
3. Square pitch equal to 1.158 times pin diameter
    (See Figure 4)

Spacing between modules will be varied to determine the effect on reactivity.

4.2 Facility Description

The proposed critical experiments will be performed in the CX-10 critical facility at the Company's Lynchburg Research Center. CX-10 is a tank-type facility licensed for the performance of critical experiments with water moderated UO_2 and mixed oxide lattices. The available core tanks consist of a 5 foot diameter aluminum tank within a 9 foot diameter steel tank. In order to accommodate 25 modules at wide spacing, it will be necessary to remove the smaller tank and construct all loadings inside the 9-foot diameter tank.

A detailed description of the facility is presented in references 1-11.

4.3 Fuel Rods

The fuel to be used in this program consists of 2.46% enriched UO_2 pellets clad in aluminum. Use of this fuel in the CX-10 facility was authorized in amendment #5. It has been in use at the reactor since 1962.

Table 1 summarizes the physical properties of the 7000 fuel rods. These rods are the property of DOE and are presently located at B&W's Critical Facility. The uncertainties shown in Table 1 are standard deviations of the means obtained from vendor's quality control data and check measurements on 50 to 100 randomly selected samples. The impurities are given as the summation of $N_i \sigma_i$ where N_i is the concentration of each impurity per cubic centimeter of the oxide fuel, and σ_i is the corresponding microscopic absorption cross section at 2200 m/sec. The end caps are 0.3-cm-thick aluminum plugs. A 2.5 cm-long dead space between the top of the fuel pellet stack and the top end cap is filled with Kaowool.
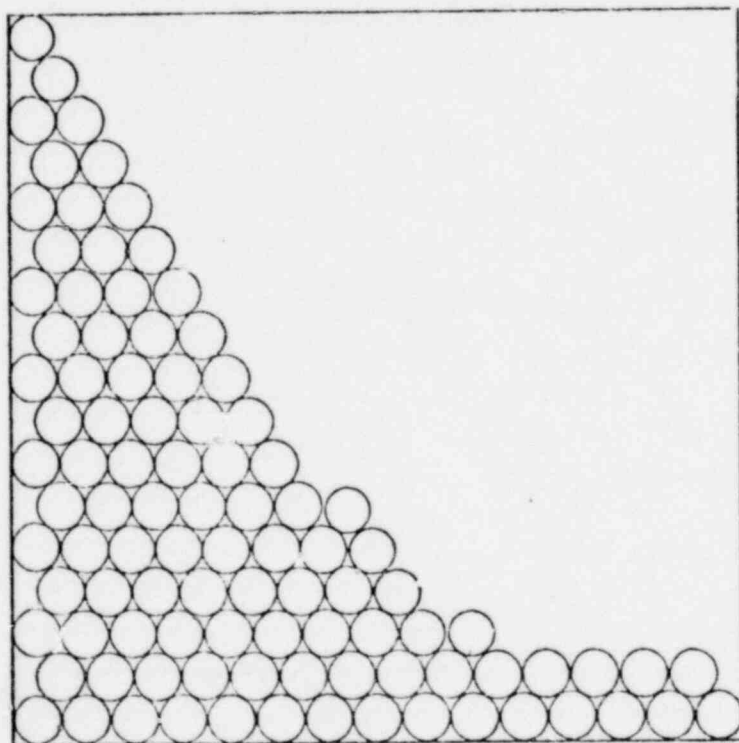
FIGURE 1

TWENTY-FIVE MODULES LOADED IN CORE TANK

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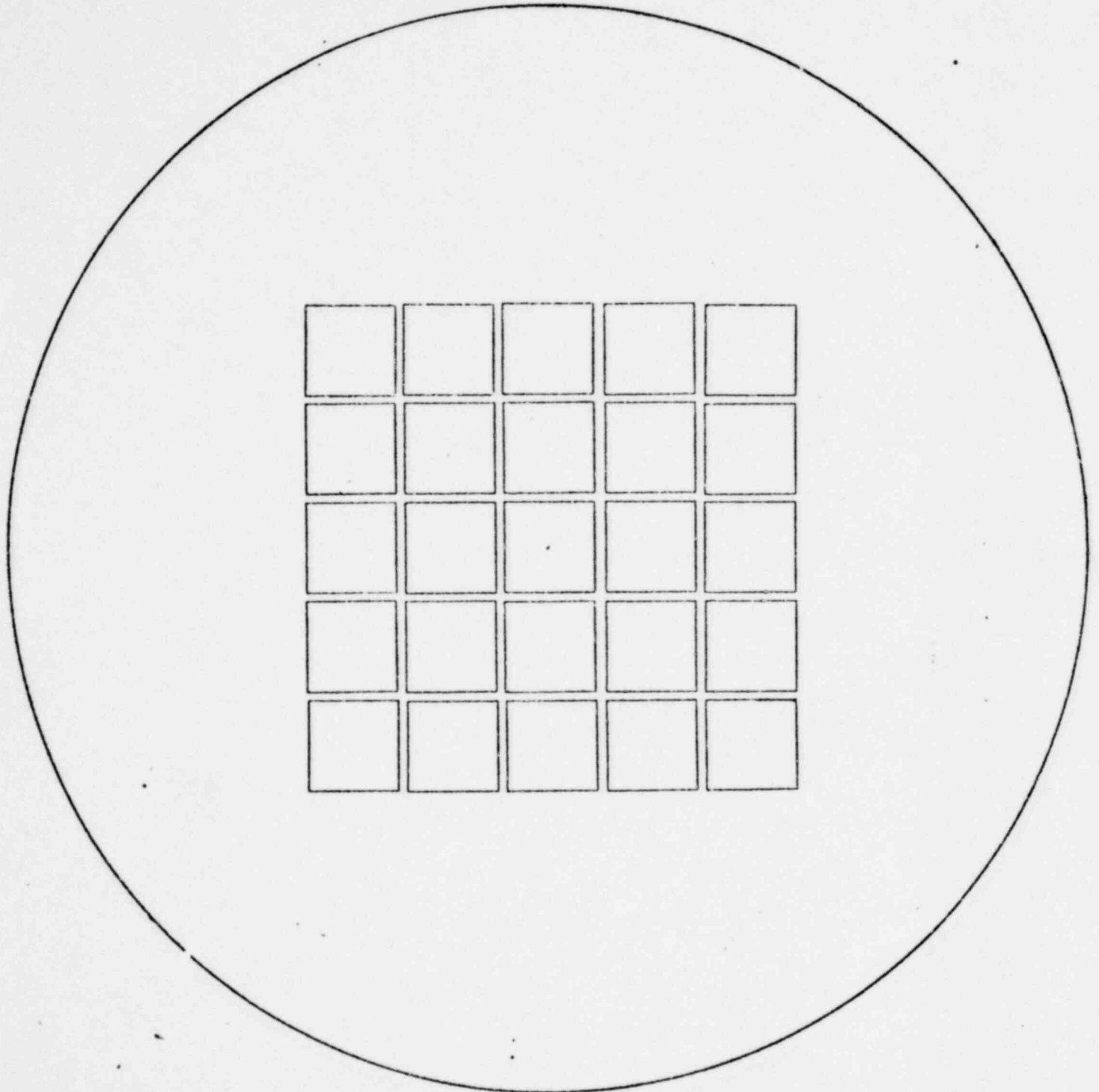
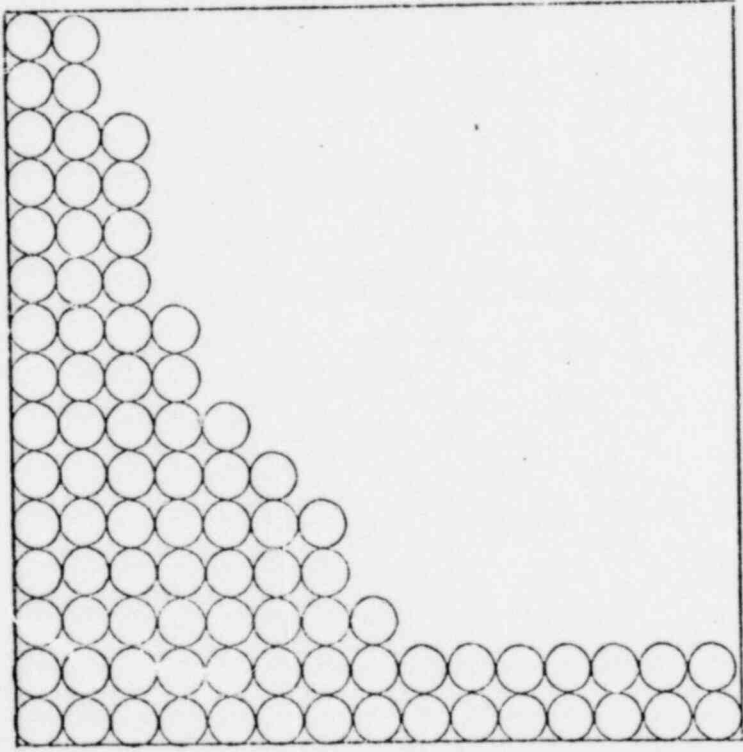FIGURE 2

MODULE WITH FUEL RODS

TOUCHING ON TRIANGULAR PITCH
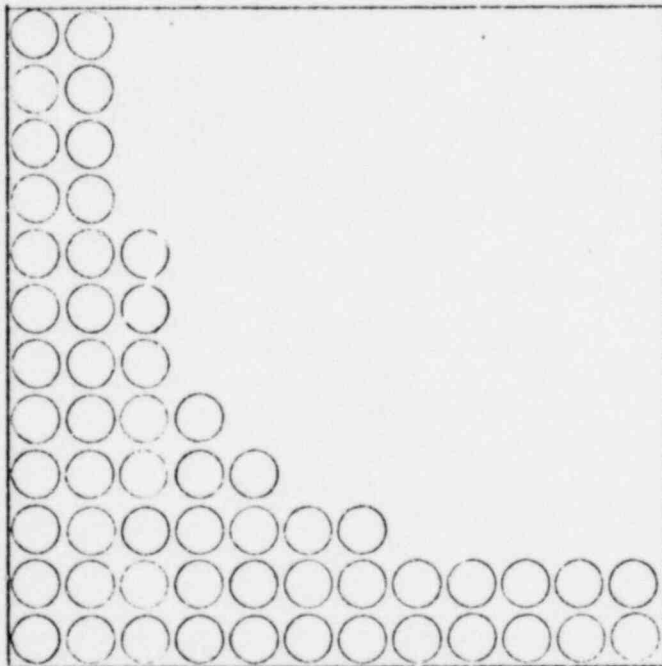

NON-MODERATOR TO MODERATOR RATIO = 9.7

FIGURE 3

MODULE WITH FUEL RODS

TOUCHING ON SQUARE PITCH

NON-MODERATOR TO MODERATOR RATIO = 3.7

FIGURE 4
MODULE WITH FUEL RODS
SLIGHTLY SEPARATED ON SQUARE PITCH

NON-MODERATOR TO MODERATOR RATIO = 1.4
(With Pitch of 0.550 inches)

Table 1. Properties of 2.46% Enriched UO_2 Fuel Rods

Parameter	Average	$\sigma(x)$	$\sigma(\bar{x})$
OD, cm	1.206	0.002	0.0003
Wall thickness, cm	0.081	--	0.003
Wall material	6061-T6 Al	--	--
Pellet diameter, cm	1.030	0.001	0.0005
Total length, cm	156.44	0.41	0.05
Active fuel length, cm	153.34	0.88	0.02
Pellet length, cm	1.914	0.008	0.001
Wt of UO_2, g/rod	1305.5	39.7	1.0
Wt U/wt UO_2, %	88.13	0.01	0.00
Wt of U, g/rod	1150.5	35.0	0.9
Wt of ^{235}U, g/rod	28.29	0.86	0.02
Enrichment, wt 25/wt U, %	2.459	0.002	0.001
Pellet density, g/cm^3	10.29	0.05	0.02
Bulk density, g/cm^3	10.22	0.36	0.01
$\Sigma N_i \sigma_i$ (summation of impurities), cm^2/cm^3 oxide	$<4 \times 10^{-4}$	--	--

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4.4 Experiment Preparation

Calculation of anticipated critical condition will be performed for each case prior to the start of the experimental program. These calculations will include an evaluation of control blade worth. They will be performed by the nuclear criticality safety group at the Lynchburg Research Center using state-of-the-art monte carlo and transport criticality codes.

Before the experimental phase of the program starts, the detailed experiment design and program plan will be reviewed and approved by the Company's Safety Review Committee. The charter and make-up of the Safety Review Committee is described in Reference 12.

4.5 Measurements

4.5.1 General

B&W will load a total of 5 to 9 benchmark cores and attempt to bring each one critical. Each core will simulate LWR fuel that has been disassembled and stored in tightly-packed cannisters. Twenty-five clusters of close-packed fuel pins will make-up each benchmark core. To ensure that a critical condition can be achieved at an acceptable moderator height, boric acid will be added to the moderator as needed. Reactivity control will be provided by a combination of moderator boron addition (or removal) and water level adjustment. All 4 safety blades will normally be fully withdrawn from the core, and will perform a safety function only (i.e., blades will not normally be used for reactivity control).

All operations will be performed by NRC licensed personnel in accordance with the current CX-10 license and established procedures. Experimental verification of rod worth will be made at low power when the initial criticality of each core is achieved. The power calibration for each loading will be determined from the activation of standardized gold foils.

The current schedule calls for the experiments to start in February of 1980 and to last about 12 months.

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4.5.2 Fuel Assemblies

The fuel pins will be loaded into assemblies (clusters) that are about 7 inches square. The framework for the assemblies will be provided by top and bottom end fittings fastened together by 4 threaded aluminum rods (one at each of the 4 corner positions). Two continuous aluminum straps about one-third of a pin length from the top and bottom of each fuel bundle will surround the fuel rods to maintain the lattice spacing between the end fittings. Lattice spacing in the loose-packed, square pitch fuel bundles will be maintained by top, bottom, and center grid plates.

A specially designed aluminum plate will replace the continuous grid at the bottom of the core tank. In the same way that the bottom grid was use to position individual fuel rods, the base plate will position the fuel bundles. Tie plates attached at the top end fitt of contiguous assemblies will fasten the core together in monolithic fashion. Additional hardware will rigidly brace the core against the core tank walls.

4.5.3 Triangular Pitch Loadings

The triangular pitch clusters will consist of a 15x17 matrix of fuel rods loaded on a triangular pitch equal to the fuel pin diameter (Figure 2). Three cores will be constructed from 25 of these triangular pitch modules. Core 1 will be loaded with about the minimum spacing between modules at which criticality can be reached. Core 3 will be loaded with about the maximum spacing between modules at which criticality can be reached. Core 2 will be loaded at an intermediate spacing

4.5.4 Square Pitch Loadings

Two different fuel clusters with pins loaded on a square pitch will be assembled. In the first, a 15x15 matrix of fuel rods will be loaded on a square pitch equal to the pin diameter. In the second, a 13x13 matrix of fuel rods will be loaded on a square pitch equal to 1.158 times the pin diameter. Loadings constructed from these clusters will make up an additional 2 to 6 benchmark cores.

4.6 Health Physics

Health physics procedures and practices will comply with Appendix H of Reference 12. A copy of reference 12 is kept in the CX-10 control room at all times.

4.7 Emergency Procedures

In the event of a site emergency, administrative control will be assumed by the emergency organization as described in reference 13. A copy of reference 13 is maintained in the CX-10 control room at all times.

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5.0 EVALUATION OF MAXIMUM CREDIBLE ACCIDENT

Discussions in Section VIII of reference 4 of the potential accidents listed below apply to the proposed program:

1) Operation at excessive power

2) Mechanical failures

3) Accidental flooding of dry core

4) Continued control rod withdrawal

5) Continued addition of moderator and addition of incorrect moderator, including loading errors

Although the core design of the planned experimental program differs in two respects from that discussed in reference 4 (i.e., top grid has been eliminated and the use of a core tank cover is not contemplated), the same rationale applies in the consideration of potential accidents. The top end fittings are bolted over the fuel rods. " fuel bundles are fastened together by tie plates that are bolted down into place. Specially designed hardware located between the top edge of the core and the inside of the core tank will rigidly fix the core assembly into vertical position. It is inconceivable that the tie plates or end fittings could be lifted up to cause the spacing between fuel rods or bundles to change during operation. We therefore conclude, as was concluded in reference 4, that no serious, credible threat to safe operation is posed by the first three potential accidents listed above. Items 4 and 5 are considered in greater detail in the following sections.

5.1 Continued Moderator Addition

As in references 4 and 6, the worst moderator addition accident is based on continuous addition of unborated H_2O to a core intended to require a high moderator boron concentration. The probability of this type of accident is extremely remote because:

1. Experiments with unborated moderator will be done infrequently and will usually be scheduled as a group.

2. Changes in moderator boron concentration can be made
    only with the approval of the senior operator in
    charge, and must be recorded in the operational log-
    book. Since moderator changes require considerable
    time and effort in connecting hoses, pumps, etc., an
    unauthorized moderator change would probably be
    noticed by members of the experimental group.

3. A check of the logbook is made before each run to
    verify moderator boron concentration.

4. The operators are trained to rely on instrument
    response rather than anticipated critical conditions.
    Any dramatic increase in reactivity at low moderator
    heights should be observed by the operator before
    criticality is reached.

Even though this sequence of events is extremely unlikely, the result is a
maximum reactivity addition rate of only .047%/sec, compared to the 0.2%/sec
calculated in references 4 and 6. As discussed in the following section, this
addition rate is slightly less than the .05%/sec ramp associated with continued
rod withdrawal.

5.2 Continued Rod Withdrawal

The license limit on reactivity addition by rod withdrawal is .05%/sec.
It is unlikely that reactivity would ever be added at this rate because:

1. All rods are usually cocked before moderator addition
    begins and are not usually used for reactor control.

2. At full moderator height, a maximum reactivity addi-
    tion rate of .05%/sec corresponds to a total worth
    of 3% and withdrawal rate of 30 in/minute (reference 6).
    The worth of any individual safety blade, measured when initial
    criticality is reached, is kept below 3%. In addition,
    the normal rod withdrawal rate is slower than 30
    in/minute.

3. The operator would have to ignore all instrument
indications that the reactor is supercritical and
continue safety blade withdrawal.

The resulting .05%/sec ramp is slightly greater than the .047%/sec
ramp calculated for continued addition of incorrect moderator. A reactivity
addition of .05%/sec to a just critical core is therefore selected as the
maximum credible accident.

5.3 Analysis of Maximum Credible Accident

5.3.1 Analytical Model

To determine which of the proposed cores would be the most sensitive to
a ramp insertion of reactivity, the neutron lifetime, average fuel temperature
coefficient of reactivity, and hot channel factor for each case was calculated using
the monte carlo criticality core KENO IV and the XSDRN neutron transport code
(Appendices A-2 & A-3). The results are listed in Table 2.

Table 2. Nuclear Parameters of Tightly-Packed UO_2 Lattices

	average α_f, $10^{-5}/^\circ C$	neutron lifetime, sec.	hot channel factor
Tight-packed square pitch module, 0.7" spacing	-2.59	2.96×10^{-5}	7.0
Loose-packed square pitch module, 0.7" spacing	-1.54	3.15×10^{-5}	5.8
Triangular pitch module, 0.8" spacing	-2.16	3.98×10^{-5}	7.4
Triangular pitch module, 1.5"	-1.65	3.98×10^{-5}	7.8

On the basis of these results, the second and fourth cores listed above were chosen
for analysis.

The nuclear excursion that would result from a .05%/sec reactivity addition
rate was analyzed using an adiabatic point kinetics model with Doppler feedback.
The same point kinetics model described in reference 4 was used. The analyses

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in reference 4 and 6 show that the adiabatic assumption is a good one for the temperatures and timeframe involved in this analysis. The equations that comprised the analytical model were:

$$\rho = Bt + \alpha_f(T_F - T_{Fo})$$

$$k = 1/(1 - \rho)$$

$$\frac{dP}{dt} = \frac{k(1 - \beta) - 1}{\ell} P + \frac{k}{\ell} \sum \lambda_i C_i'$$

$$\frac{dC_i'}{dt} = -\lambda_i C_i' + \beta_i P$$

$$\frac{dT_F}{dt} = \frac{P}{C_F M_F}$$

where

ρ = reactivity ($\rho_o = 0$)

B = reactivity addition rate

α_f = effective fuel temperature coefficient

T_F = fuel temperature

T_{Fo} = initial fuel temperature

k = neutron multiplication factor

P = reactor power

ℓ = neutron lifetime

β = effective delayed fraction

M_F = fuel mass

C_F = heat capacity of fuel

The subscript "i" refers to the ith delayed neutron precursor group. The value for each parameter used in the analysis is listed in Table 3.

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Table 3. Parameters Used In Kinetics Calculations

Parameter	Value
Water Height	150 cm
β_{eff}	.0075
ℓ, 10^{-5} sec	3.15 (3.98 for triangular pitch core)
Average α_f, $10^{-5}/^\circ C$	-1.54 (-1.65 for triangular pitch core)
Initial Power, watts	10^{-3}
Initial Temp., $^\circ C$	22
Fuel Mass, gm	5.4×10^6 (7.9×10^6 for triangular pitch core)
Clad Mass	9.1% of fuel mass
Fuel Heat Capacity, $\dfrac{joules}{gm\ ^\circ C}$	0.277
Clad Heat Capacity, $\dfrac{joules}{gm\ ^\circ C}$	0.941

5.3.2 Results

The calculations show an initial power surge that is quickly terminated by Doppler broadening of the absorption resonances in the heated fuel. The safety blades are assumed to be inserted 1.2 sec after 1.5 kW (max. scram setting) is reached. This occurs after the initial power surge has passed, and well before a significant quantity of heat can be conducted into the cladding. This is the same response time assumed in reference 6. Finally, 6 seconds after rod drop (7.2 seconds after scram signal is initiated) the power is assumed to drop to zero as moderator begins to drain from the core tank. A plot of core power versus time is shown in Figure 5.

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Figure 5. Core Power Vs Time
(.05%/sec ramp, loose square)

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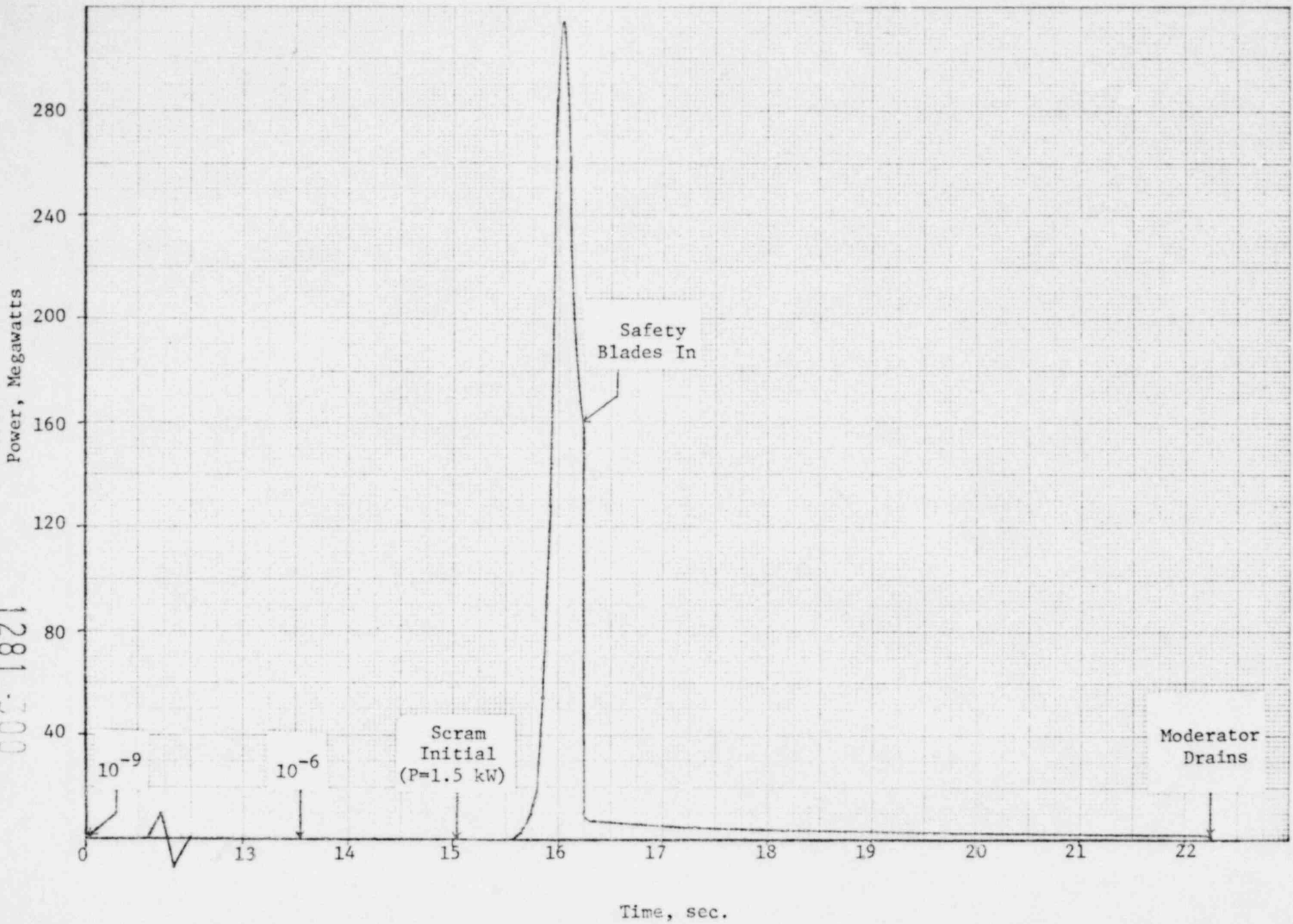
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The cladding hot spot temperature was calculated as discussed in Appendix A-3. It was assumed that all of the energy generated during the excursion is retained in the clad and fuel (i.e., moderator drains from the core before heat can be conducted into it).

These assumptions are conservative because:

1. There are four independent safety channels, two period and two power level. In practice, one of the two redundant period channels would sense the excursion and terminate it long before the prompt critical condition is reached. The proper operation of every safety channel is verified with a radiation source each day that a run is made.

2. No credit is taken for reactivity insertion as the rods are falling. In addition, the sum of instrument response time and rod drop time is substantially less than 1.2 sec.

3. Total rod worth will always exceed the 2% assumed in the analysis.

4. The ratio of maximum to average temperature in the fuel is 5.8 to 7.8, since it is derived from the power distribution. However, because of the much higher thermal conductivity of the cladding in comparison to the fuel, the ratio of maximum to average cladding temperature will be much less at short times.

5. During the period between rod drop and water drop, heat will be transferred into the water from the fuel-clad system, thereby lowering the final temperature.

The results of the point kinetics analyses are given in Table 4. Table 4 also provides a comparison between these results and the results for the MCA considered in reference 6. From this comparison, we conclude that the consequences of the maximum credible accident are no more severe than those associated with past critical experiment programs, even if partial failure of the safety system is assumed.

Table 4. Results of MCA Analysis

	P_o, W	Peak Power Fuel Mass, watts/gm	Min. Period, sec.	T_{max} of Fuel at Rod Insertion, °F	T_{max} of Clad, °F	Total Energy Released, MW-sec
Loose Square, 0.7" Separation	10^{-3}	58.1	.063	639.2	611.3	101.2
Triangular Pitch, 1.5" Separation	10^{-3}	57.7	.067	686.8	663.1	120.4
Amend 5 Case (0.2%/sec ramp)	10^{-3}	294	.024	1200	1050	75

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APPENDIX A

A-1. CALCULATION OF MAXIMUM REACTIVITY ADDITION RATE BY MODERATOR FILL

The first step in computing the maximum reactivity addition rate is to determine the maximum excess reactivity possessed by each core at full water height with unborated moderator. These calculations were performed on the Company's CDC-7600 computer with the monte carlo criticality code. KENO IV. The 123-group XSDRN cross section set was used. The results are tabulated below.

Table A-1. Maximum Excess Reactivities

Loading	Max K_{ex}	Max ρ_{ex}
Loose-packed square pitch modules	0.23	18.7%
Tight-packed square pitch modules	0.14	12.3%
Triangular pitch modules	.09	8.26%

Past experimental measurements with the 2½% fuel show that for moderator heights below 135 cm, the following equation expresses the relation between reactivity and water height:

$$\frac{\delta\rho}{\delta H} = \frac{C}{(H+\lambda)^3} \qquad \text{Eq. (1)}$$

where

H = water height

ρ = reactivity

C & λ are fitting constants

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Recent measurements at the facility indicate that $C = 8.225 \times 10^6$ ¢/cm^2 and $\lambda = 9.26$ cm. The excess reactivity between some height H_o and 135 cm is therefore

$$\rho_{ex} \quad \% \ \frac{\Delta K}{K} \ = \ \frac{-C/2}{(H+\lambda)^2} \ \Bigg|_{H_o}^{135}$$

$$= \frac{30,844}{H_o + 9.26} - 1.482 \qquad\qquad \text{Eq. (2)}$$

The reactivity associated with an increase in moderator height from 135 to 145 cm is about 0.18%. From Eq. (2) and the values of maximum excess reactivity calculated earlier, the minimum critical water heights for each core were determined.

Assuming a maximum fill rate of 40 gpm, equation (1) was used to compute the reactivity addition rate at each critical water height. The results of these calculations are tabulated below.

Table A-2. Maximum Reactivity Addition Rates

Loading	Minimum Critical Water Height (Accident Condition)	$\frac{\delta\rho}{\delta t}$ @ minimum Critical Water Height
Loose-packed square pitch modules	30 cm	.047%/sec
Tight-packed square pitch modules	38.4 cm	.0275%/sec
Triangular pitch modules	47.5 cm	.0162%/sec

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A-2. CALCULATION OF SELECTED NUCLEAR PARAMETERS

A.2.1 Neutron Lifetime

The neutron lifetime for each core was computed with the monte carlo criticality code KENO IV. The 123-group XSDRN cross section set was used. KENO calculates the neutron lifetime as follows:

$$TIML = \frac{\sum\limits_{NBA} \left[\sum\limits_{NPB} \frac{AB*TME + WT*TME}{FNPB} \right]}{FNBA}$$

where

AB = absorption weight

WT = leakage weight

TME = elapsed time

FNPB = NPB = number of neutrons per generation

FNBA = NBA = number of generations

In words, the lifetime calculated by KENO is the average time that a neutron spends in the system befc leakage or absorption. The standard deviation from statistical uncertainty was about 1% for each case. Since the neutron lifetime varies with K_{eff}, the lifetime calculation was made for systems with a neutron multiplication of about unity.

A.2.2 Fuel Temperature Coefficient of Reactivity

The fuel temperature coefficient was calculated by first computing the reactivity of a cold (22°C) unborated core. With the cladding and water held at 22°C, the reactivity of the same core with a uniform fuel temperature rise of 1000°F was calculated. The average fuel temperature coefficient was obtained by dividing the calculated reactivity change by 1000°F. This is the fuel temperature coefficient of reactivity reported in Table 2.

Reactivity of each core was calculated with the XSDRN transport code using a 1-D model. Homogenized cross sections developed from the AMPX computer package were used.

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In the real case, the fuel temperature rise across the core will not be uniform. To account for this effect, a cosine power distribution and $\cos^2$ importance function for the heated fuel was assumed. On this basis, it was estimated that the effective fuel temperature change was a factor of 2.37 greater than the average fuel temperature change. For conservatism, this factor was assumed to have a value of 2.0. Accordingly, the effective fuel temperature coefficient (for the purposes of the point kinetics model) was assumed to be $2.0 \times \text{average} \alpha_f$.

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A-3 CALCULATION OF PEAK-TO-AVERAGE POWER

The peak-to-average (PTA) power ratio was needed to calculate the maximum fuel and clad temperatures. The variation in power from fuel cluster to fuel cluster was estimated by assuming a cosine power distribution in all three directions. This resulted in a PTA power ratio of 3.88 for the fuel assemblies.

Because the core is made up of tightly packed fuel lattices separated by water gaps, however, large variations in power from pin-to-pin may occur within a single fuel assembly. To determine the PTA power ratio for the core, therefore, the PTA pin power within a fuel assembly was multiplied by the PTA fuel assembly power.

The PTA pin power within each of the 3 assembly types was determined from a one dimensional XSDRN "super-cell" calculation. Each assembly was modeled as a homogeneous cell, surrounded by a water layer equivalent to the water gap width separating the assemblies. The PTA pin power within an assembly was calculated directly from the power distribution inside a homogenized cell, assuming that the cell was located within an infinite array of such homogenized assemblies. The 25 group cross section set for the super-cell calculation was obtained by collapsing the 123 group cross section set from the AMPX computer package. The results of these calculations are summarized in Table A-7.

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Table A-3. Hot Channel Factors For Various Cores

Assembly Type	Water Gap Thickness, in.	*Calculated Peak-to-Average Pin Power In Assembly	Core Hot Channel Factor (PTA pin powerx3.88)
Tight square	0.7	1.56 (1.8)	7
Loose square	0.7	1.3 (1.5)	5.8
Triangular pitch	0.8	1.71 (1.9)	7.4
Triangular pitch	1.5	1.81 (2.0)	7.8

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*For conservatism, the values listed in parentheses were used to estimate hot channel factor.

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7. BAW-1023, "Application for Amendment No. 6 to Critical Experiment License No. CX-10 to Permit Use of Materials In-Core," R. M. Ball, B&W Co., February 1965. (Supplemented by Letters to AEC from B&W dated March 29 and May 21, 1965).

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9. Letter to USNRC from A. F. Olsen (B&W Co.), dated April 30, 1976.

10. Letter to B&W Co. from D. L. Ziemann (USNRC), dated March 2, 1977.

11. Letter to USNRC from A. F. Olsen (B&W Co.), dated June 26, 1978.

12. BAW-381, "Demonstration and Condition for License SNM-778," B&W Co., May 1970 (revised).

13. "Lynchburg Research Center Emergency Plan," B&W Co., December, 1978. (Forwarded with letter to USNRC from A. F. Olsen dated Jan. 29, 1979)

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