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ORNL-4169/V2/S1

The Oak Ridge Research Reactor—Safety Analysis

Volume 2
Supplement 1

F. T. Binford

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THE OAK RIDGE RESEARCH REACTOR - SAFETY ANALYSIS

Volume 2
Supplement 1

F. T. Binford

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THE OAK RIDGE RESEARCH REACTOR - SAFETY ANALYSIS

F. T. Binford

A. INTRODUCTION

In order to provide adequate bases for technical specifications it has been found desirable to expand the heat transfer analysis originally presented in the ORR Safety Analysis (ORNL 4169, Vol. II). Moreover, the treatment in that document tacitly assumed that the reactor fuel used in the ORR would be of the well-known U-Al alloy plate-type. A large amount of experience with this fuel in conditions similar to and more rigorous than those encountered in the ORR was available, and so there was no reason to include an extensive analysis of its limitations in ORNL 4169. It is now proposed to fabricate ORR fuel using the U_3O_8-Al dispersion technique which has been successfully used in the HFIR since 1965.[1,2] The experience with HFIR has demonstrated conclusively that the U_3O_8-Al elements are superior to the U-Al fuel elements in virtually every respect. However, since it is also proposed to increase the ^{235}U content above 265 grams per element, which is the maximum weight previously used in ORR, it appears prudent to examine the implications of the change to oxide fuel.

REFERENCES

1. R. W. Knight, J. Binns, G. M. Adamson, Jr., *Fabrication Procedure for Fabricating HFIR Fuel Elements*, ORNL 4242 (1968).
2. F. T. Binford, R. W. Knight, "The Use of U_3O_8-Al Cermet Fuel in Research Reactors," *Trans. Am. Nuc. Soc.* 27, pp 834-5 (Nov. 1977).

B. THE ESTABLISHMENT OF OPERATING PARAMETERS ASSOCIATED WITH HEAT TRANSFER

Part I - Development of the Method

1. INTRODUCTION

The operation of the ORR is conducted with restrictions imposed on certain of the operating parameters. These restrictions are more than adequate to prevent overheating of the reactor fuel, and therefore serve to guarantee the integrity of the fuel cladding. The selection of limiting values of these parameters is based upon a number of calculations,[1,2,3] the results of which have been supported by tests and measurements in the reactor.[4,5,6,7]

The relevant parameters are heat flux, coolant velocity, and subcooling. None of these quantities can be directly measured; however, they may be inferred from a knowledge of the measurable parameters: reactor power, primary coolant flow, coolant temperatures, and pressure. The establishment of a functional relationship among these quantities requires an understanding of the hydraulic conditions which exist in the core configuration under consideration, together with a knowledge of the spatial distribution of the heat flux.

The problem was investigated by Webster[1] and others, and it is the purpose of this study, which is largely an extension of Webster's work, to summarize the methods to be followed in establishing appropriate ranges of values for the significant operating parameters.

2. ESTIMATION OF THE CRITICAL HEAT FLUX

The critical or "burnout" heat flux may be roughly defined as that value of the heat flux at which, under the conditions considered, the temperature of a heated surface begins to increase at a rapid and uncontrolled rate.

There have been a number of empirical correlations developed to describe the relation between critical heat flux, coolant velocity, and degree of subcooling. Webster, in his work, determined that the correlation most suitable for the conditions which prevail in the ORR is that of Labuntsov.[1,8]

The data upon which this correlation is based fall within the range of pressure, heat flux, and subcooling characteristic of ORR operation.

$$\phi_c = 4.61 \times 10^5 \Theta(p)[1 + \frac{0.232v^2}{\Theta(p)}]^{1/4} [1 + \frac{15C_p \Delta t_{sub}}{\lambda(p)^{1/2}}] \qquad (1)$$

where

ϕ_c = critical heat flux (Btu/ft^{2}hr)

C_p = specific heat of coolant (Btu/lb°F)

p = pressure (atm. abs.)

λ = heat of vaporization (Btu/lb)

v = coolant velocity (ft/sec)

p_c = critical pressure of coolant (atm. abs.)

$\Theta(p) = p^{1/3}(1-p/p_c)^{4/3}$

t_{sat} = saturation temperature (°F)

t_i = inlet temp (°F)

Δt = coolant temperature (°F)

$$\Delta t_{sub} = t_{sat} - t_i - \Delta t = \text{subcooling (°F)}$$

The pressure p and the corresponding saturation temperature and heat of vaporization will, of course, depend upon the pressure drop through the reactor, and this in turn depends upon the coolant velocity. Although the pressure may be as high as 40 psia at the site of maximum heat flux, the pressure has, both for simplicity and conservatism, been chosen to be 1.633 atm. (24 psia - the core exit pressure) for all the cases considered. Moreover, the location of maximum heat flux (which actually occurs about 9 in. above the core exit) will be assumed to occur at the core exit. These assumptions will result in an underestimate of the critical heat flux because they imply the minimum possible subcooling. Based on these assumptions, $p = 1.633$ atm. abs., $\Theta(p) \simeq 1.166$, $t_{sat} = 238$°F, $\lambda = 953$ Btu/lb, $C_p = 1$ Btu/lb°F, $p_c = 218$ atm. abs. Thus, Eq. (1) becomes

$$\phi_c = 5.38 \times 10^5(1 + 0.199v^2)^{1/4}(1 + 1.232 \times 10^{-2}\Delta t_{sub}) \qquad (2)$$

In order to obtain a relationship between ϕ_c and v it is necessary to specify Δt_{sub} as a function of v. Normally the core inlet temperature, t_i, is about 120°F. A conservative value of 125°F will, however, be used. Hence

$238 - 125 - \Delta t$

$$\Delta t_{sub} = 113 - \Delta t \quad .$$

A Flow area ~ 4.173 × 10⁻² ft²

The mass flow rate through an ORR fuel element is nominally $9.24 \times 10^3 v$ lb/hr.[4] It has been found[1,9] that the ratio of maximum to average heat flux in the axial direction is 1.44. The heat transfer area of an ORR fuel element is 14.96 ft^2, so if ϕ_{max} is the maximum heat flux in the reactor, the heat input to the hottest element becomes $14.96\phi_{max}/1.44 = 10.38\phi_{max}$ Btu/hr. Thus, for the hottest channel

$$\Delta t = 1.123 \times 10^{-3}\phi_{max}/v \qquad (°F)$$

$\frac{qA}{wc_p}$

The width of an individual fuel channel may be only 90% of the nominal width, hence the temperature rise is increased by the corresponding amount and

$$\Delta t = 1.248 \times 10^{-3}\phi_{max}/v \qquad (°F) \tag{3}$$

Thus

$$\Delta t_{sub} = 113 - 1.248 \times 10^{-3}\phi_{max}/v \tag{4}$$

Setting $\phi_{max} = \phi_c$ and putting this result into Eq. (2) we obtain

$$\phi_c = 5.38 \times 10^5(1 + 0.199v^2)^{1/4}(2.392 - 1.538 \times 10^{-5}\phi c/v)$$

$v = 28$ ft/sec

which may be solved for ϕ_c to yield

$\phi_c = 2.2 \times 10^6$

$$\phi_c = \frac{1.287 \times 10^6 v(1 + 0.199v^2)^{1/4}}{v + 8.279(1 + 0.199v^2)^{1/4}} \qquad (\text{Btu/ft}^2\text{hr}) \tag{5}$$

This relation, which gives the critical heat flux in an ORR core as a function of coolant velocity, holds for heat fluxes between 4×10^5 and 1.5×10^7 Btu/ft^{2}hr for pressures above one atmosphere, and for values of Δt_{sub} greater than zero. It contains the following explicit conservative assumptions:

(i) The maximum heat flux occurs at the point of lowest pressure and highest coolant temperature.

(ii) The width of the hottest channel is 90% of the nominal width, thus reducing the mass flow rate thereby 10%.

(iii) The inlet temperature is 125°F.

3. THE ONSET OF NUCLEATE BOILING

The power level at which the onset of boiling is observed was determined as a function of primary coolant flow in a series of experiments

performed in the ORR.[6] For flows in the turbulent region (above about 1500 gpm) it was found that

$$Q_b = 0.0244\ F_T^{0.8} \tag{6}$$

where Q_b is the power level (MW) at which boiling begins when the total core flow is F_T (gpm). These tests were performed using a core containing 26 fuel elements and 4 shim-safety rods. The coolant inlet temperature was 120°F.

4. ESTIMATION OF THE "BURNOUT" POWER LEVEL

In order to express Eq. (5) in terms of directly observable quantities, it is necessary to establish a relationship between the maximum heat flux ϕ_m and power level Q and between total primary coolant flow F_T and the coolant velocity v. Both of these depend upon the core configuration.

For a given configuration the average velocity through a fuel channel is proportional to the flow through the reactor core. It may be deduced from data obtained during a series of hydraulic tests[5] that the average velocity through a fuel element may be related to the pressure drop through the core by the equation

$$v = 5.0\Delta p^{0.56}$$

Because the velocity in an outer fuel channel may be only 85% of the average,[4] the relationship to be used is

$$v = 4.25\ \Delta p^{0.56} \tag{7}$$

During a recent series of experiments[10] it was found that the flow through the reactor core F_c may be expressed as a function of the pressure drop by the relation

$$F_c = 94.1\ n\ \Delta p^{0.56} \tag{8}$$

where n is the number of core elements and F_c is the primary coolant flow through the core expressed in gallons per minute. Thus

$$v = 4.52 \times 10^{-2} \frac{F_c}{n} \tag{9}$$

The heat transfer area of a fuel element is 14.96 ft^2 and that of a shim-safety rod is 10.80 ft^2. The shim rods are assumed halfway withdrawn so the total heat transfer area becomes $14.96n_f + 5.4n_s$, where n_f is the number of fuel elements and n_s the number of shim-safety rods. The average

heat flux therefore becomes

$$\overline{\phi} = 3.413 \times 10^6 Q/(14.96 n_f + 5.4 n_s) \qquad (\mathrm{Btu/ft^2hr}) \qquad (10)$$

where Q is the reactor thermal power in MW.

Let K be the ratio of maximum to average heat flux. Then the peak heat flux is $K\overline{\phi}$. Upon setting $K\overline{\phi} = 3.413 \times 10^6 KQ/(14.96 n_f + 5.4 n_s)$ equal to the critical heat flux in Eq. (5) and upon using Eq. (9) and (10) we obtain

$$\frac{KQ_c}{2.77 n_f + n_s} = \frac{1.112 \times 10^{-2}\beta(1 + 4.059 \times 10^{-4}\beta^2)^{1/4}}{5.46 \times 10^{-3}\beta + (1 + 4.059 \times 10^{-4}\beta^2)^{1/4}} \qquad (11)$$

where here for convenience we have set

$$\beta \equiv F_c/n, \qquad (12)$$

the nominal average flow through an ORR core element, and Q_c is the power level which will produce ϕ_c somewhere in the core.

The geometrical and velocity uncertainties have already been accounted for in this formulation. An additional hot channel factor of 1.1 is included to take care of other contingencies. Hence, the minimum burnout power is estimated by the relation

psi

$$\frac{KQ_{bo}}{2.77 n_f + n_s} = \frac{1.01 \times 10^{-2}\beta(1 + 4.059 \times 10^{-4}\beta^2)^{1/4}}{5.46 \times 10^{-3}\beta + (1 + 4.059 \times 10^{-4}\beta^2)^{1/4}} = \Psi(\beta) \qquad (13)$$

It should be noted that two more conservative assumptions have been incorporated:

(iv) The maximum heat flux occurs in an outside channel where the velocity is only 85% of the average.

(v) The burnout power has been arbitrarily reduced by a factor of 0.91 to cover other contingencies.

Another element of conservatism which exists, but which is not specifically included, is the fact that during operation at high power some of the heat (1 or 2%) is lost to the reactor pool water and does not contribute to the primary coolant temperature rise. For this reason, the Δt values calculated in this report are slightly higher than those actually observed.

Values of $\psi(\beta)$ have been tabulated as a function of β and are given in Table 1.

Table 1. Values of $\psi(\beta)$

β	$\psi(\beta)$	β	$\psi(\beta)$
10	9.58200E-2	510	2.7595
20	0.182778	520	2.80081
30	0.263128	530	2.84186
40	0.333723	540	2.88268
50	0.410856	550	2.92325
60	0.480363	560	2.96359
70	0.547776	570	3.00371
80	0.613445	580	3.0436
90	0.677609	590	3.08327
100	0.740439	600	3.12272
110	0.802067	610	3.16196
120	0.862595	620	3.20099
130	0.922108	630	3.23982
140	0.980676	640	3.27845
150	1.03836	650	3.31688
160	1.09521	660	3.35511
170	1.15128	670	3.39316
180	1.2066	680	3.43102
190	1.26122	690	3.46869
200	1.31517	700	3.50618
210	1.36847	710	3.54349
220	1.42117	720	3.58063
230	1.47327	730	3.61759
240	1.52481	740	3.65438
250	1.57581	750	3.69101
260	1.62629	760	3.72747
270	1.67626	770	3.76376
280	1.72576	780	3.7999
290	1.77478	790	3.83588
300	1.82336	800	3.8717
310	1.8715	810	3.90737
320	1.91921	820	3.94288
330	1.96652	830	3.97825
340	2.01343	840	4.01347
350	2.05995	850	4.04855
360	2.1061	860	4.08348
370	2.15189	870	4.11827
380	2.19732	880	4.15292
390	2.2424	890	4.18743
400	2.28715	900	4.22181
410	2.33158	910	4.25605
420	2.37569	920	4.29016
430	2.41948	930	4.32414
440	2.46297	940	4.35799
450	2.50617	950	4.39172
460	2.54908	960	4.42531
470	2.5917	970	4.45879
480	2.63405	980	4.49214
490	2.67613	990	4.52537
500	2.71795	1000	4.55847

Some of the total primary coolant flow, F_T, is diverted to demineralizers, to cool experimental facilities, and for other purposes. This bypass flow, F_B, has been found[10] to follow the relation

$$F_B = 293 + 5.65\ \Delta p^{0.98} \tag{14}$$

Since $F_T = F_C + F_B$, the relationship between the total (measured) primary flow and the flow through the core, $n\beta = F_C$, can be found by combining Eq. 7, 8, 9, and 14, so that

$$F_T = 293 + \beta\ (n + 1.987 \times 10^{-3}\beta^{0.75}) \tag{15}$$

Values of β versus F_T for several values of n are given in Table 2*.

The value of K, the peak to average heat flux ratio, remains to be determined. In general, this ratio will depend upon the core configuration and must be determined for each loading.

To a first approximation K will be equal to K′, the ratio of peak to average fissioning density. Actually, the value should be somewhat less than this because not all of the energy is deposited at the fission site and some of the heat is conducted in directions normal to the fuel-coolant interface. (The total area available for heat transfer in a standard ORR fuel element is 18.77 ft^2 of which 14.96 ft^2 or 80% directly covers fuel.) Examination of a number of core loadings[11] produces the following values of K′ for cores containing more than 24 core elements:

$$K' = 4.47 - 0.2n + 0.0046n^2 \tag{16}$$

as a function of n, the number of core elements. Estimates of the actual peak to average heat flux which occurred during the boiling experiments[6] using a method developed by Bergles and Rohsenow[12] lead to values about two-thirds of this. Hence K will be taken to be 80% of K′, that is

$$K = 3.58 - 0.16n + 0.0037n^2 \tag{17}$$

It must be emphasized that this holds only for "normal" core configurations, and that core loadings which contain experimental configurations which may significantly skew the heat flux must be investigated on a case by case basis.

---

*A useful approximation in the range of interest is $\beta \simeq 0.948F_T n^{-0.99} - 8$.

Table 2. Values of F_T (gpm) as A Function of β and n

β	n					
	25	27	29	31	33	35
0	293	293	293	293	293	293
10	543.112	563.112	583.112	603.112	623.112	643.112
20	793.376	833.376	873.376	913.376	953.376	993.376
30	1043.76	1103.76	1163.76	1223.76	1283.76	1343.76
40	1294.26	1374.26	1454.26	1534.26	1614.26	1694.26
50	1544.87	1644.87	1744.87	1844.87	1944.87	2044.87
60	1795.57	1915.57	2035.57	2155.57	2275.57	2395.57
70	2046.37	2186.37	2326.37	2466.37	2606.37	2746.37
80	2297.25	2457.25	2617.25	2777.25	2937.25	3097.25
90	2548.23	2728.23	2908.23	3088.23	3268.23	3448.23
100	2799.28	2999.28	3199.28	3399.28	3599.28	3799.28
110	3050.42	3270.42	3490.42	3710.42	3930.42	4150.42
120	3301.65	3541.65	3781.65	4021.65	4261.65	4501.64
130	3552.94	3812.94	4072.94	4332.94	4592.94	4852.94
140	3804.32	4084.32	4364.32	4644.32	4924.32	5204.32
150	4055.77	4355.77	4655.77	4955.77	5255.77	5555.77
160	4307.3	4627.3	4947.3	5267.3	5587.3	5907.3
170	4558.9	4898.9	5238.9	5578.9	5918.9	6258.9
180	4810.58	5170.58	5530.58	5890.58	6250.58	6610.58
190	5062.32	5442.32	5822.32	6202.32	6582.32	6962.32
200	5314.13	5714.13	6114.13	6514.13	6914.13	7314.13
210	5566.02	5986.02	6406.02	6826.02	7246.02	7666.02
220	5817.97	6257.97	6697.97	7137.97	7577.97	8017.97
230	6069.99	6529.99	6989.99	7449.99	7909.99	8369.99
240	6322.08	6802.08	7282.08	7762.08	8242.08	8722.08
250	6574.23	7074.23	7574.23	8074.23	8574.23	9074.23
260	6826.45	7346.45	7866.45	8386.45	8906.45	9426.45
270	7078.73	7618.73	8158.73	8698.73	9238.73	9778.73
280	7331.08	7891.08	8451.08	9011.08	9571.08	10131.1
290	7583.49	8163.49	8743.49	9323.49	9903.49	10483.5
300	7835.97	8435.97	9035.97	9635.97	10236.	10836.
310	8088.51	8708.51	9328.51	9948.51	10568.5	11188.5
320	8341.11	8981.11	9621.11	10261.1	10901.1	11541.1
330	8593.77	9253.77	9913.77	10573.8	11233.8	11893.8
340	8846.49	9526.49	10206.5	10886.5	11566.5	12246.5
350	9099.28	9799.28	10499.3	11199.3	11899.3	12599.3
360	9352.12	10072.1	10792.1	11512.1	12232.1	12952.1
370	9605.02	10345.	11085.	11825.	12565.	13305.
380	9857.99	10618.	11378.	12138.	12898.	13658.
390	10111.	10891.	11671.	12451.	13231.	14011.
400	10364.1	11164.1	11964.1	12764.1	13564.1	14364.1
410	10617.2	11437.2	12257.2	13077.2	13897.2	14717.2
420	10870.4	11710.4	12550.4	13390.4	14230.4	15070.4
430	11123.7	11983.7	12843.7	13703.7	14563.7	15423.7
440	11377.	12257.	13137.	14017.	14897.	15777.
450	11630.4	12530.4	13430.4	14330.4	15230.4	16130.4
460	11883.8	12803.8	13723.8	14643.8	15563.8	16483.8
470	12137.3	13077.3	14017.3	14957.3	15897.3	16837.3
480	12390.8	13350.8	14310.8	15270.8	16230.8	17190.8
490	12644.4	13624.4	14604.4	15584.4	16564.4	17544.4
500	12898.	13898.	14898.	15898.	16898.	17898.

Table 2 - continued

β	n					
	25	27	29	31	33	35
510	13151.8	14171.8	15191.8	16211.8	17231.8	18251.8
520	13405.5	14445.5	15485.5	16525.5	17565.5	18605.5
530	13659.3	14719.3	15779.3	16839.3	17899.3	18959.3
540	13913.2	14993.2	16073.2	17153.2	18233.2	19313.2
550	14167.1	15267.1	16367.1	17467.1	18567.1	19667.1
560	14421.1	15541.1	16661.1	17781.1	18901.1	20021.1
570	14675.1	15815.1	16955.1	18095.1	19235.1	20375.1
580	14929.2	16089.2	17249.2	18409.2	19569.2	20729.2
590	15183.3	16363.3	17543.3	18723.3	19903.3	21083.3
600	15437.5	16637.5	17837.5	19037.5	20237.5	21437.5
610	15691.8	16911.8	18131.8	19351.8	20571.8	21791.8
620	15946.1	17186.1	18426.1	19666.1	20906.1	22146.1
630	16200.4	17460.4	18720.4	19980.4	21240.4	22500.4
640	16454.8	17734.8	19014.8	20294.8	21574.8	22854.8
650	16709.3	18009.3	19309.3	20609.3	21909.3	23209.3
660	16963.8	18283.8	19603.8	20923.8	22243.8	23563.8
670	17218.3	18558.3	19898.3	21238.3	22578.3	23918.3
680	17472.9	18832.9	20192.9	21552.9	22912.9	24272.9
690	17727.6	19107.6	20487.6	21867.6	23247.6	24627.6
700	17982.3	19382.3	20782.3	22182.3	23582.3	24982.3
710	18237.	19657.	21077.	22497.	23917.	25337.
720	18491.9	19931.9	21371.9	22811.9	24251.9	25691.9
730	18746.7	20206.7	21666.7	23126.7	24586.7	26046.7
740	19001.6	20481.6	21961.6	23441.6	24921.6	26401.6
750	19256.6	20756.6	22256.6	23756.6	25256.6	26756.6
760	19511.6	21031.6	22551.6	24071.6	25591.6	27111.6
770	19766.6	21306.6	22846.6	24386.6	25926.6	27466.6
780	20021.8	21581.8	23141.8	24701.8	26261.8	27821.8
790	20276.9	21856.9	23436.9	25016.9	26596.9	28176.9
800	20532.1	22132.1	23732.1	25332.1	26932.1	28532.1
810	20787.4	22407.4	24027.4	25647.4	27267.4	28887.4
820	21042.7	22682.7	24322.7	25962.7	27602.7	29242.7
830	21298.	22958.	24618.	26278.	27938.	29598.
840	21553.4	23233.4	24913.4	26593.4	28273.4	29953.4
850	21808.9	23508.9	25208.9	26908.9	28608.9	30308.9
860	22064.4	23784.4	25504.4	27224.4	28944.4	30664.4
870	22319.9	24059.9	25799.9	27539.9	29279.9	31019.9
880	22575.5	24335.5	26095.5	27855.5	29615.5	31375.5
890	22831.2	24611.2	26391.2	28171.2	29951.2	31731.2
900	23086.8	24886.8	26686.8	28486.8	30286.8	32086.8
910	23342.6	25162.6	26982.6	28802.6	30622.6	32442.6
920	23598.4	25438.4	27278.4	29118.4	30958.4	32798.4
930	23854.2	25714.2	27574.2	29434.2	31294.2	33154.2
940	24110.1	25990.1	27870.1	29750.1	31630.1	33510.1
950	24366.	26266.	28166.	30066.	31966.	33866.
960	24622.	26542.	28462.	30382.	32302.	34222.
970	24878.	26818.	28758.	30698.	32638.	34578.
980	25134.1	27094.1	29054.1	31014.1	32974.1	34934.1
990	25390.2	27370.2	29350.2	31330.2	33310.2	35290.2
1000	25646.3	27646.3	29646.3	31646.3	33646.3	35646.3

As an example of the use of these methods, consider a 30-element core which contains 26 fuel elements and 4 shim-safety rods and for which the total coolant flow is 17,300 gpm. From Table 2, β = 563.2 gpm. $2.77n_f + n_s$ = 76.02 and K = 2.11. From Table 1, ψ (563.2) = 2.9764, hence Q_{bo} = 107 MW which is somewhat lower than Webster's original estimate of 138 MW. The difference may be attributed to the conservatism added by assuming that the peak heat flux occurs at the core exit.

It should be noted that in the range under consideration the burnout power decreases as the number of core elements is increased, e.g., for 17,300 gpm and a 35-element core Q_{bo} = 91 MW. This is because the combination of increase in peaking factor and decrease in velocity has a greater effect than the decrease in average heat flux.

5. THE POWER LEVEL AT THE ONSET OF NUCLEATE BOILING

Equation (6) can be modified in a manner similar to that set forth above in order to accommodate cores other than the 30-element core to which the original data applied. Since the data are experimental, the appropriate peaking factor is already present in the 30-MW case. Thus

$$\frac{KQ_{nb}}{2.77n_f + n_s} = 1.029 \times 10^{-2}\beta^{0.8} \tag{18}$$

A word of caution is in order here: because Eq. (6) is based on direct measurements of flow and power at the observed onset of boiling, the heat flux peaking factor K, whatever its value, is already present. Clearly the use of a value of K higher than that which, in fact, existed in order to estimate the heat flux at the onset of boiling would predict a higher value than is actually required to cause boiling. Thus, Eq. (18) should not be used to estimate the boiling heat flux unless the peaking factor is well known. Equation (18) retains the variation in K with the number of fuel elements, but its absolute value is based upon the observed behavior of a 30-element core.

PART II - APPLICATION

1. INTRODUCTION

The ORR is normally operated at a power level of 30 MW with a total primary coolant flow of about 18,000 gpm and an inlet temperature of 120°F. Although the number of core elements may vary depending upon the experiment requirements, the reactor usually contains 6 shim-safety rods and about 25 fuel elements. Under these conditions, the temperature rise through the core* is 11.8°F, and the average heat flux is 2.52×10^5 Btu/ft^{2}hr. The temperature rise through the hottest channel is 21.4°F and the peak heat flux is about 5.5×10^5 Btu/ft^{2}hr.

Using the conservative methods outlined above, the critical heat flux for this flow and configuration is about 2.39×10^6 Btu/ft^{2}hr and corresponds to a burnout power of about 104 MW. The onset of nucleate boiling would be expected to occur at about 59 MW.

Because the ORR is a research tool, it is necessary to retain as much flexibility as possible in the operation. Thus at times it is desirable to operate the reactor at power levels other than 30 MW and at flows other than 18,000 gpm. Moreover, the number of fuel elements may be varied. A listing calculated by the methods of Part I of the values of the parameters of importance to safety for three "normal" configurations and for various primary system flows is presented in Table 3. It should be noted that in some cases of low flow, the results of Eq. (13) imply a negative subcooling in the hot channel, a condition for which the Labuntsov equation (Eq. 1) is invalid. In these cases, the critical heat flux and corresponding burnout power level has been estimated for the velocities under consideration by using Eq. (1) and requiring a non-negative subcooling at the channel exit.

2. REACTOR PROTECTION

The reactor is equipped with a number of devices which can be preset to initiate protective action when they detect that one of the parameters which they are designed to measure has exceeded the predetermined limits established for this parameter. The parameters which are utilized to control the

*Neglects heat loss to reactor pool.

Table 3. Thermal-Hydraulic Conditions in ORR Cores at Burnout and at the Onset of Nucleate Boiling

	Primary Coolant Flow, GPM	Hot Channel Coolant Velocity, ft/sec	Burnout Heat Power, MW	Hot Channel ΔT at Burnout, °F	Core ΔT at Burnout, °F	Hot Channel Exit Temperature at Burnout, °F	Core Exit Temperature at Burnout, °F	Incipient Boiling Power Level, MW	Hot Channel ΔT at Incipient Boiling, °F	Core ΔT at Incipient Boiling, °F	Hot Channel Exit Temperature at Incipient Boiling, °F	Core Exit Temperature at Incipient Boiling, °F	Core Pressure Drop, psia
29 Fuel Elements 6 Shim-Safties	2,500	2.84	14.00	113	43.9	238	168.9	9.72	78.5	30.5	203.5	155.5	0.49
	5,000	6.06	29.83	113	43.9	238	168.9	17.81	67.5	26.2	192.5	151.2	1.88
	7,500	9.27	45.63	113	43.9	238	168.9	25.02	62.0	24.1	187.0	149.2	4.03
	10,000	12.48	58.66	108.0	42.0	233	167.0	31.74	58.4	22.7	183.4	147.7	6.84
	12,500	15.68	70.33	103.0	40.0	228	165.0	38.10	55.8	21.7	180.8	146.7	10.29
	15,000	18.88	81.43	99.0	38.5	224	163.5	44.20	53.7	20.9	178.7	145.9	14.34
	17,500	22.07	91.77	95.4	37.1	220.4	162.1	50.09	52.1	20.2	177.1	145.7	18.95
	20,000	25.17	101.74	92.4	35.9	217.4	160.9	55.80	50.7	19.7	175.7	144.7	24.12
	22,500	28.45	111.32	89.8	34.9	214.8	159.9	61.37	49.5	19.2	174.5	144.2	29.82
25 Fuel Elements 4 Shim-Safties	2,500	3.21	15.90	113	49.9	238	174.9	10.78	76.6	33.8	201.6	158.8	0.61
	5,000	6.83	53.84	113	49.9	238	174.9	19.75	65.9	29.1	130.9	154.1	2.64
	7,500	10.45	51.24	111.8	49.4	236.8	174.4	27.64	60.3	26.6	185.3	151.6	5.38
	10,000	14.08	65.00	105.5	46.6	230.5	171.6	35.18	57.1	25.2	182.1	150.2	8.49
	12,500	17.68	77.80	100.2	44.3	225.2	169.3	42.22	54.4	24.0	179.4	149.0	12.76
	15,000	21.29	89.84	96.1	42.5	221.1	167.5	48.98	52.4	23.1	177.4	148.1	17.77
	17,500	24.88	101.25	92.8	41.0	217.8	166.0	55.50	50.9	22.5	175.9	147.5	23.48
	20,000	28.48	112.14	89.9	39.7	214.9	164.7	61.82	49.6	21.9	174.6	146.9	30.73
	22,500	32.07	122.56	87.2	38.5	212.2	163.5	67.97	48.4	21.4	173.4	146.4	36.92
21 Fuel Elements 4 Shim-Safties	2,500	3.98	18.74	113	58.9	238	183.9	12.16	73.4	38.2	198.4	163.2	0.89
	5,000	8.47	39.87	113	58.9	238	183.9	22.25	63.1	32.9	188.1	157.9	3.43
	7,500	12.95	57.77	107.0	55.8	232.0	180.8	31.25	57.9	30.2	182.9	155.2	7.31
	10,000	17.42	75.01	100.3	52.4	225.3	177.8	39.62	54.4	28.4	179.4	153.4	12.41
	12,500	21.87	87.13	95.9	49.8	220.9	174.8	47.54	52.3	27.2	177.3	152.2	18.65
	15,000	26.32	100.35	91.5	47.7	216.5	172.7	55.13	50.3	26.2	175.3	151.2	25.95
	17,500	30.76	112.84	88.0	45.9	213.0	170.9	62.45	48.7	25.4	173.7	150.4	34.28
	20,000	35.19	124.73	85.1	44.3	210.1	169.3	69.54	47.4	24.7	172.4	149.7	43.59
	22,500	39.61	136.08	82.5	42.9	207.5	167.9	76.45	46.3	24.1	171.3	149.1	53.84

reactor are coolant flow, inlet temperature, exit temperature, coolant temperature rise (Δt), inlet pressure, exit pressure, core pressure drop (Δp), and, indirectly, reactor power. The reactor power is determined calorimetrically by comparing the coolant flow and the temperature rise across the core. This value is used to calibrate a radiation detection instrument which in turn provides a direct indication of power level. Because of its rapid response, the radiation detection instrument is used to directly initiate protective action when it detects an abnormal power increase.

In addition, protective action is automatically initiated by the instruments which measure coolant flow, the core exit temperature, core ΔT and core ΔP. During startup protective action is also initiated by radiation detection instruments which measure the rate of power increase.

There are several levels of protective action which may occur. These are related to the magnitude of the anomaly detected and are as follows:

(a) "Alarms," which alert the operator by means of an audible and visible signal.

(b) "Set-backs," which produce an automatic reduction of power by means of the servo-system until either the condition causing the set-back has cleared or the reactor power has been reduced to 1% of full power.

(c) "Reverses," which are automatic motor-driven insertions of all the shim-safety rods to shut the reactor down.

(d) "Scrams," which cause the shim-safety rods to be released and to fall into the reactor under gravity, thus initiating an almost instantaneous shutdown.

The sequence of these actions is that given above, although an alarm always accompanies the other three actions. The instrumentation provides the protection actions described above. However, alarms, set-backs, and reverses are initiated by the "control" system, whereas "fast scrams" originate in the reactor "safety" system. The latter system is designed with adequate redundancy to provide maximum reliability and is completely independent of the control system. Its function is to seize control of the reactor and immediately shut it down should this become necessary.

Although there have been numerous instances of automatic control system action, the safety system has never been required to scram the reactor

during normal power operation because a reactor scram trip point was actually exceeded.*

3. THE ESTABLISHMENT OF OPERATING PARAMETERS

The remainder of this part is devoted to developing a rationale for establishing limitations on the operating parameters which guarantee safe and orderly operation, but which at the same time do not inhibit the flexibility which is required of a research reactor such as the ORR.

The parameters of importance are coolant velocity, heat flux, and subcooling in the hot channel. However, as was pointed out previously, none of these can be directly measured. Thus, it becomes necessary to infer these values from the measurable bulk temperatures, pressures, flows, and the derived power level using a scheme such as that proposed in Part I.

The general philosophy followed is to first establish "safety limits" for these parameters which delineate a region within which there is confidence that no serious consequences (in this case loss of fuel cladding integrity) can occur. Second "limiting safety system settings" (LSSS) are established. These limits define a region within which the operation can be carried on in an orderly fashion, which is bounded by scram trip points, and which is within the safety limit region such that the scrams will prevent any safety limit from being exceeded. Third, control system settings (CSS) are established which provide for alarms, set-backs, reverses, and slow scrams and which are designed to prevent the necessity for safety system action. These define the region in which normal operations are carried on, and this region is contained within that defined by the LSSS.

This philosophy is generally consistent with the DOE approach;[13] however, in that case no credit is taken for the control system protective actions. Although the control system has proven to be a formidable protective device to prevent parameter excursions up to and beyond the safety scram trip points, it will not be considered further here. Nevertheless, the protective actions of the control system will be continued in operation

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*In some cases, experimental rigs are tied into the safety system to scram the reactor in order to prevent damage to the experiment should it, itself, malfunction. There have been one or two scrams for this cause.

at the present or at a higher state of reliability. Because of this, it is expected that the safety system trip points will seldom, if ever, be reached. To accomplish this, some "head room" is required between the LSSS and the normal operating level in order to accommodate the control system set points.

The three important parameters, primary coolant flow (F_T), reactor thermal power (Q), and core temperature rise (ΔT), are physically related such that

$$\Delta T \cong 6.92 \times 10^3 \; Q/F_T \qquad (°F) \tag{19}$$

Thus, if a limiting relationship is established between any two of these quantities this relationship defines the value of the third. In establishing such a relationship, two conditions are of interest: those conditions which could result in burnout and those conditions which represent the onset of incipient boiling. Curves showing the relationship between total primary system flow and thermal power level for these two conditions for three core configurations are shown in Fig. 1. The corresponding values of ΔT for both the core and the hot channel are given in Table 3.

The Safety Limit corresponding to these curves should be set somewhere below the burnout conditions since it is required that the burnout be prevented. Once safety limits on flow and power are established, the corresponding safety limit on core ΔT follows automatically as a result of the relationship given in Eq. (19).

With respect to the flow-power safety limits, it is suggested that the following criterion is adequate for "normal" cores which contain between 25 and 35 core elements (21-31 fuel elements, 4-6 shim-safety rods). The flow-power safety limit shall consist of a curve for which the power level is 0.925, that calculated for the burnout of a 35-element core. Because of the conservative assumption that the maximum heat flux occurs at the core exit rather than near the core center where the saturation temperature is higher, the actual margin provided is neared to 1.3 than the 1.08 margin implied and is judged adequate. The Safety Limit curve is shown as Curve A in Fig. 2 and the data, including the ΔT safety limits, are given in Table 4.

Based upon an inlet temperature of 125°F, the core exit temperature safety limits are just 125° + ΔT, and these values are also given in Table 4.

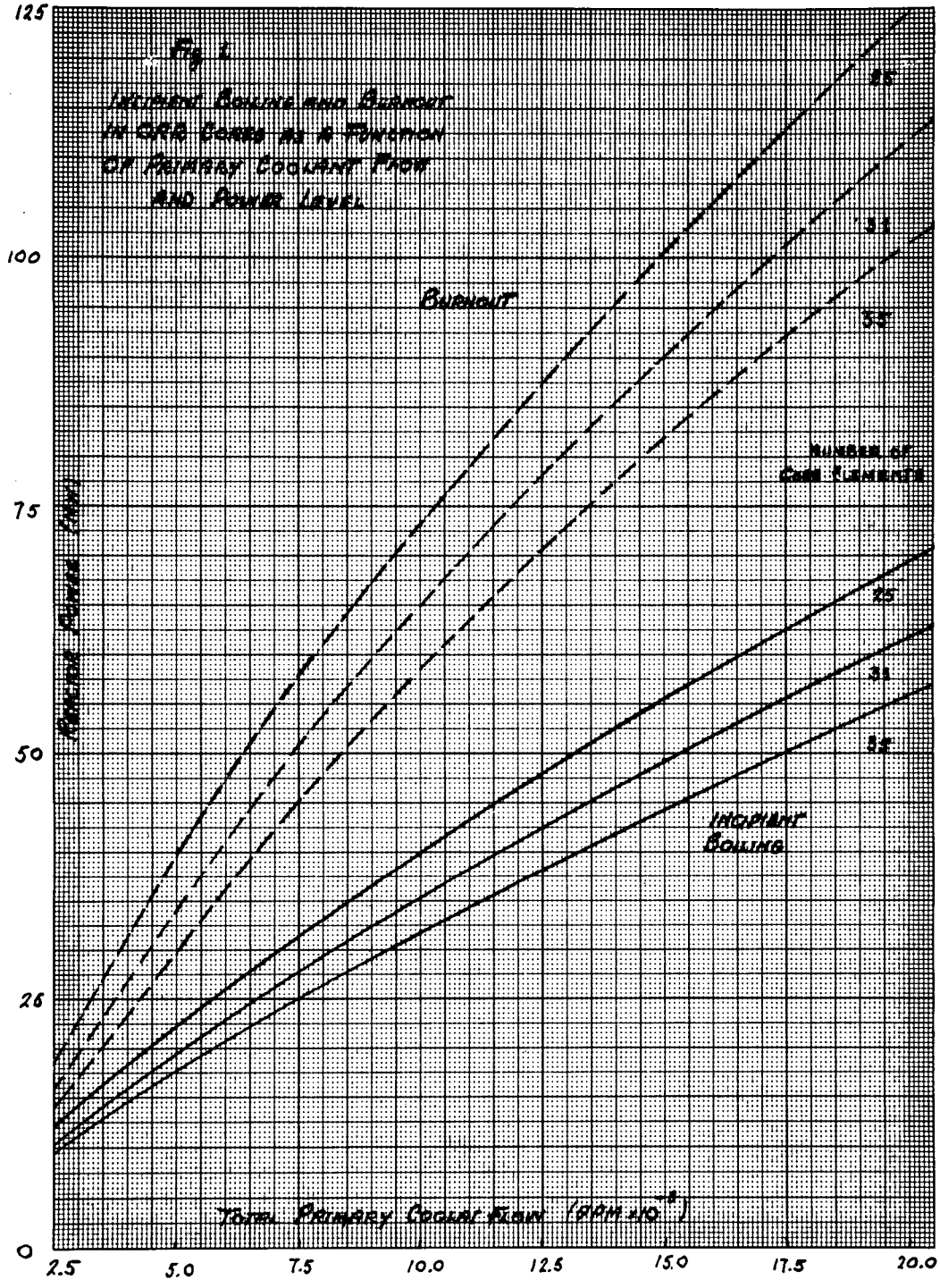
Fig. 2
Incipient Boiling and Burnout in CRB Cores as a Function of Primary Coolant Flow and Power Level
Burnout
Incipient Boiling
Number of Core Elements
25
31
35
Power Level (MW)
Total Primary Coolant Flow (gpm x10^-3)
0
25
50
75
100
125
2.5
5.0
7.5
10.0
12.5
15.0
17.5
20.0

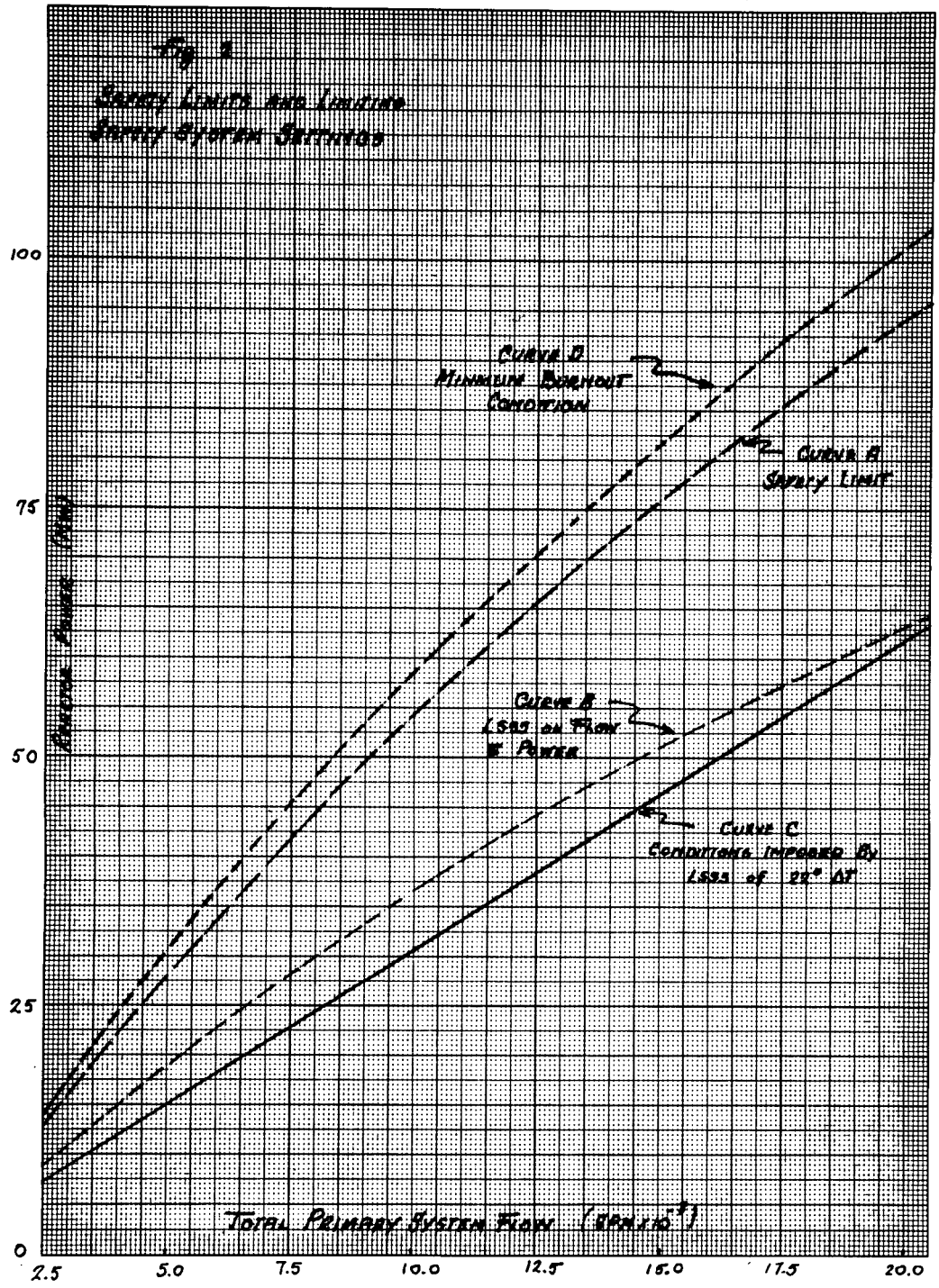
Fig. 2
Safety Limits and Indicate
Safety System Settings
Curve D
Minimum Burnout
Condition
Curve A
Safety Limit
Curve B
Loss on Flow
& Power
Curve C
Conditions Imposed By
Loss of 22° ΔT
Reactor Power (MW)
Total Primary System Flow (GPM x 10⁻³)
100
75
50
25
0
2.5
5.0
7.5
10.0
12.5
15.0
17.5
20.0

Table 4. Safety Limits and Limiting Safety System Settings on Flow Power and ΔT

F_T (gpm)	Q_{SL} (MW)	ΔT_{SL} (°F)	Q_{LSSS} (MW)	ΔT_{LSSS} (°F)	Q_{LSSS}* (MW)
2,500	12.9	40.6	8.7	27.4	7.0
5,000	27.6	40.6	18.6	27.4	15.0
7,500	42.2	40.6	28.5	27.4	22.8
10,000	54.3	38.9	36.6	26.2	30.8
12,500	65.0	37.0	43.9	25.0	38.6
15,000	75.3	35.6	50.8	24.0	46.5
17,500	84.9	34.3	57.3	23.2	54.6
20,000	94.1	33.2	63.5	22.4	62.3
22,500	103.0	32.3	69.5	21.8	70.3

*Implied value based on ΔT = 22°F.

The limiting safety system settings on flow and power are chosen to provide a margin of 1.6 below the calculated 35-element burnout curve. This is adequate to prevent any anticipated transient from exceeding the safety limits. The LSSS on flow and power are shown as Curve B in Fig. 2 and the data are given in Table 4. A corresponding set of ΔT and core exit temperature LSSS could also be established; however, in this case it is convenient to establish a single value of ΔT for all flows and power levels. A value of 22°F has been selected. This results in a power-flow curve which is always below that dictated by the flow-power LSSS and is, therefore, more restrictive. The curve implied by the 22°F ΔT is shown as Curve C in Fig. 2 and the data are presented in Table 4. The corresponding LSSS for core exit temperature is 147°F.

Safety limits and LSSS values may also be placed upon core pressure drop. Since the pressure drop is a strong function of the number of elements in the core, these must be determined experimentally for each core configuration.[10] Basically the ΔP measurement is a confirmation of the flow measurement and its value lies in its ability to detect anomalies such as bypassed flow or malfunction of the flow measuring device.

Because the inlet temperature does not give a direct measure of the conditions in the reactor, it is not used as a parameter to effect safety

system action. It is, however, an important control variable. It is important because the core exit temperature is the sum $t_{ex} = t_{in} + \Delta T$. Thus, a lowered inlet temperature could result in a higher ΔT even though t_{ex} were below the LSSS. In fact, the foregoing calculations were performed using a value of 125°F for t_{in} whereas the reactor is normally operated at a t_{in} of 120°F. This was done because the higher inlet temperature results in a lower (more conservative) value for the critical heat flux. Since, as implied above, a value of ΔT exceeding the LSSS must be accompanied by either a flow or power level anomaly, and since the subcooling is controlled by the fixed exit temperature LSSS of 147°F, there is ample protection against lowered inlet temperature.

If it is desired to correct for an inlet temperature of 120°F, it is simply necessary to reduce the core exit temperature safety limit and LSSS values by 5°F, while leaving the ΔT's unchanged. This increases the conservatism in the flow-power level curves by about 6%.

As a practical matter the safety-system trip points should be set below the LSSS. The actual values of the settings are governed by two factors: (1) the settings should be determined with regard to the accuracy of the measuring instrument to make sure that the parameters measured are indeed below the LSSS; (2) the trip points should be as far below the LSSS as is practical to permit orderly operation at the power level contemplated; however, sufficient headroom between the scram trip points and the operating level should be provided in order to permit the use of the control system protective devices.

4. CONCLUSION

The foregoing scheme for establishing safety limits and limiting safety system settings is believed to provide an adequate and extremely conservative procedure for all "normal" ORR cores. It must, however, be emphasized that in every case where this procedure is applied it must be verified that the heat flux peaking factor and the coolant velocity indeed satisfy, or are more favorable than, the values indicated by the relations developed above. That is, cores which depart significantly in configuration from those used in the past and which provided the experimental information used for the basis of the foregoing analysis must be analyzed on a case-by-case basis.

Moreover, as additional information concerning the relationships among the measurable and deduced parameters becomes available, an improved and more realistic set of values for the safety limits and LSSS may be developed.

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C. THE USE OF U_3O_8 CERMET FUEL

1. INTRODUCTION

The original ORR fuel elements were of the alloy type and were assembled by a brazing technique which was at that time (circa 1958) the only method available. Later the assembly procedure was improved and the roll-swaging method currently in use was initiated. The HFIR fuel elements first produced in the early 1960's were fabricated using U_3O_8 cores. This method was chosen because the new technique was available and because it was found that this procedure permitted better process control and produced a fuel element which was superior in quality to that produced by the alloy process.

Since August 1965 we have used about 80,000 U_3O_8 plates in the HFIR with outstanding success. During this period, on only two occasions has a fuel element developed a fission-product leak or suspected leak. In one case the problem was diagnosed as minor surface contamination, and the element was operated to depletion. In the other case the element was removed after 1500 MWD operation,[1] primarily as a concession to conservatism.

Despite the excellent performance of U_3O_8 elements in the HFIR, no change was made in the ORR elements. This was because there was no economic incentive to do so and because the alloy elements had given very satisfactory service in the somewhat less rigorous ORR environment. Since its startup in April 1958 about 40,000 alloy plates have been used in the ORR. During this period, there have been four fission-product leaks or suspected leaks. All of these elements were identified and discarded prior to depletion. In only one case, the meltdown which occurred on July 1, 1963,[2] were there any significant consequences. This incident, which was caused by an almost totally blocked fuel element, resulted in the partial melting of two of the fuel plates as the power level was being raised from 12 to 24 MW.

A comparison of the physical characteristics of the two types of plates reveals that the 6061 Al cladding of the oxide plate has about 2 1/2 times the ultimate strength of the 1100 Al cladding of the alloy plate at temperatures between 200 and 300°F. The oxide core is not as strong but is more malleable than the alloy core. This is a distinct advantage in

manufacture because it eliminated the "dog bone" problem* and because the resultant uniform cladding thickness together with the greater strength of the 6061 Al cladding actually produces a stronger element and one which is less likely to leak.

2. GENERAL CONSIDERATIONS

The U_3O_8 currently used to fabricate fuel is a high-fired oxide which is calcined at 1350 to 1400°C. The material is supplied at -100 +325 mesh with 25% fines and is blended with aluminum powder to yield the required ^{235}U content. The blended powder is compacted at approximately 22.5 TSI, vacuum degassed, and then inserted into an aluminum picture frame. The aluminum covers are welded on and the assembly is hot rolled to produce the fuel plate.

The dimensions and configuration of the oxide plates can be made identical to those of fuel plates made by the alloy process. Hence, there are no basic differences in the overall thermal performance relative to heat removal at the surface. Because of the better process control which can be exercised during the manufacture of oxide fuel, these fuel cores are more homogeneous and are less likely to exhibit significant areas of non-bonding at the fuel-cladding interface than are alloy cores.

McElroy[3] has measured values of the thermal conductivity for several U_3O_8-Al cores, and these are presented in Table 1. The thermal conductivity of several U-Al alloys is shown in Fig. 1.

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*Dog boneing" is the thickening of the fuel core at the ends during the plate rolling operation. This results in the thinning of the clad at that point. The problem is particularly bothersome when high U alloy is used.

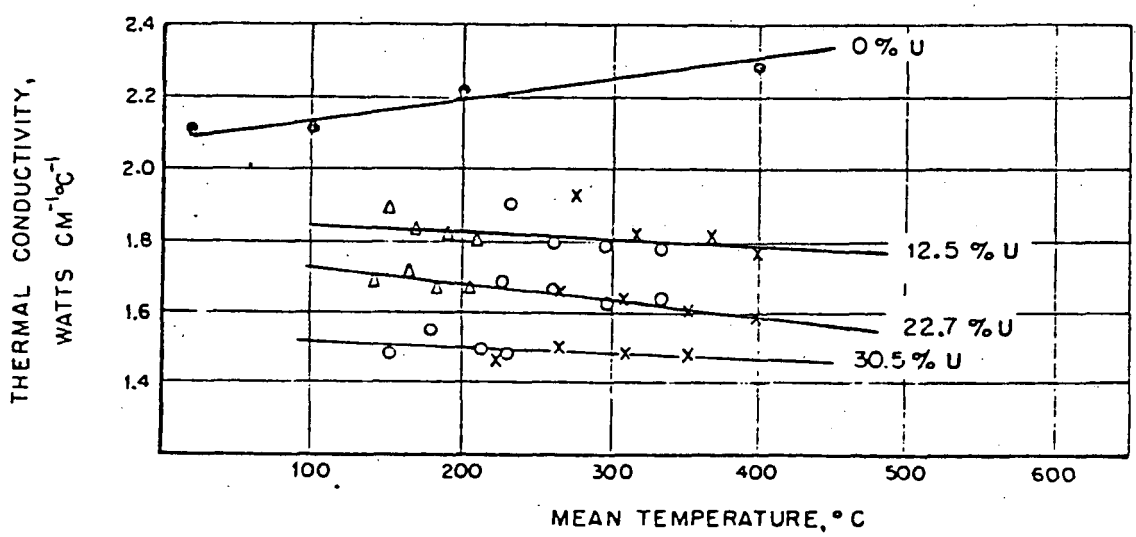
Table 1. Thermal Characteristics of U_3O_8-Al Fuel Cores
U_3O_8-Al Cermet Cores

Volume % U_3O_8	Weight % U_3O_8	Cermet Density[a] g/cm^3	g U per cm^3	K[a] W/cm°C	C[b] Joules/g°C
		Burned Oxide			
13.45	31.91	3.203	0.865	1.72	0.72
19.49	43.26	3.424	1.254	1.45	0.60
		Hi-Fired Oxide			
12.45	30.82	3.316	0.865	1.77	0.73
18.05	41.28	3.590	1.254	1.59	0.62

[a]Measured values.[3]

[b]Calculated based on weight fractions.

Note: 0.65 g U per cm^3 corresponds to a standard (0.051 x 6.10 x 60 cm) fuel plate core loaded to about 16 g U per plate, and 1.254 g U per cm^3 corresponds to about 23.3 g U per plate.



(Δ, O, AND X DENOTE RESULTS OBTAINED ON DIFFERENT RUNS)

Fig. 1. Thermal Conductivity of Several U-Al alloys.[4]

As can be seen, the thermal conductivity of the cermet fuel is as good as or better than that of alloy containing the same amount of uranium.

Through June 1977, 148 reactor cores have been consumed in the HFIR. They contained a total of 25,308 plates containing 30.2 wt% U_3O_8 and 54,612 plates containing 40.1 wt% U_3O_8. The performance of these plates in a thermal and hydraulic environment considerably more rigorous than that found in the ORR has been highly satisfactory. Since operation began in 1965 only one fission product leak has been detected.

3. SAFETY CONSIDERATIONS

There are three areas of interest related to safety: (a) the physical behavior of the U_3O_8 cermet under irradiation conditions; (b) the possibility of exothermic chemical reactions; and (c) the potential for fission product release under accident conditions.

(a) Physical Behavior

Experimental results obtained by Martin et al.[5], indicate that dispersions of U_3O_8 perform quite well to burnup levels of 1.5×10^{21} to 2.4×10^{21} fissions/cm^3 at irradiation temperatures of 60 to 120°C. They found that radiation-induced swelling was quite small and actually decreased with increased U_3O_8 loading. Post-irradiation examination showed no indication of blister, core-cladding separation, matrix cracking, or other types of structural defects. Operating conditions in the HFIR would imply an average burnup of about 5.3×10^{20} fissions/cm^3 with a peak of about 1.9×10^{21}. The temperatures involved range from 114 to 330°C. ? Burnup in the ORR is expected to vary from 1×10^{21} to 1.7×10^{21} fissions/cm^3.

(b) Chemical Reactions

An exothermic reaction involving U_3O_8 and Al to produce UO_2, U-Al solutions, and Al_2O_3 is thermodynamically possible. This reaction has been studied by a number of investigators.[6] It was found that no significant reaction occurs at temperatures below about 950°C. The energy release may be as high as 800 joules per gram of U_3O_8; however, high energy releases occur only when the U_3O_8 loading is of the order of 60 to 75 wt %. At 30 wt % the self-heating was found to be negligible.

In a later series of experiments performed in the TREAT facility[7] and which simulated actual accident conditions in the HFIR, the results indicated that even with 41 wt % HFIR fuel the U_3O_8-Al reaction is not an important energy source.

The highest weight fuel elements so far used in the ORR contain 265 g ^{235}U and correspond to about 28.5 wt % U_3O_8. Even at 350 g ^{235}U the wt % U_3O_8 would be well below the 41% material tested in TREAT (see Fig. 2). It may be concluded that the possibility of this reaction does not present a safety problem.

(c) Fission Product Release

There have been very few studies of the release of fission products from oxide-aluminum fuels. Creek[8] melted trace-irradiated UO_2-Al in air and obtained an average release of 5.6% of the rare gases and 0.003% of the iodine. Experiments reported by Parker and Barton[8] for U-Al alloy samples held above the melting temperature for 10-17 minutes in steam-air mixtures indicate a release of virtually all of the rare gases and from 27 to 97% of the iodines depending upon the maximum temperature achieved.

Because these two investigations were conducted under quite different conditions, it is not possible to definitely conclude that the fission product release from oxide cores will be less than that from U-Al alloy under accident conditions. However, the alloy melts at about 850°C, whereas the U_3O_8 melting point is 1250°C. In either case, the aluminum cladding melts at about 650°C.

Although, except in cases of complete flow blockage at high power or violent transients, none of these temperatures are expected to be approached, it is the disruption of the cladding which could result in the dispersal of fission products into the cooling system. This will take place only when the cladding fails, which occurs at about the same temperature in either case. Moreover, the type of events which could produce such elevated temperatures are characterized by essentially complete insulation of the fuel by film blanketing. Hence, the rate of temperature rise is of the order of several hundred degrees per second,

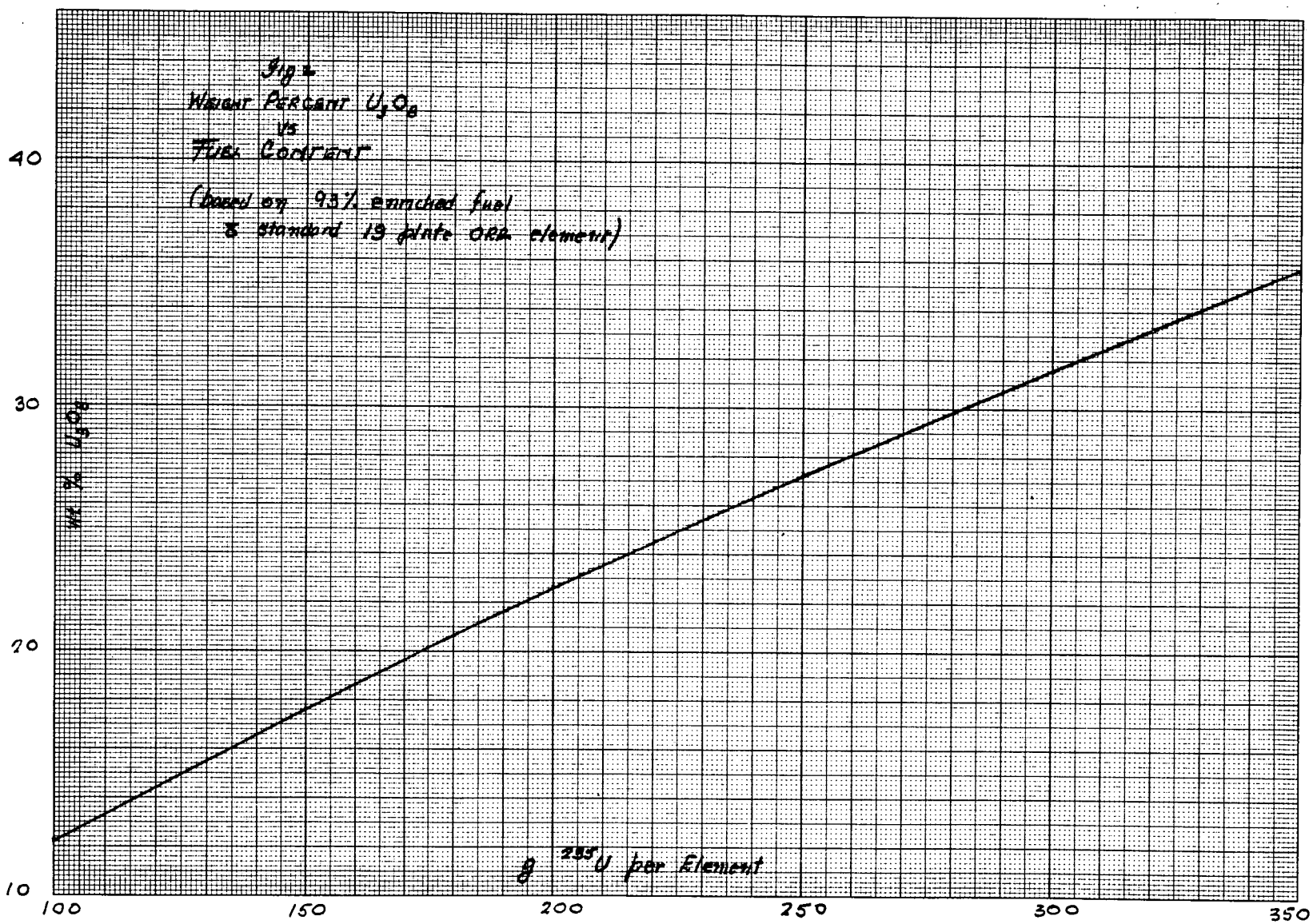


Fig 2
Weight Percent U_3O_8
vs
Fuel Content
(based on 93% enriched fuel
& standard 19 plate ORR element)
Wt % U_3O_8
g ^{235}U per Element
40
30
20
10
100
150
200
250
300
350

and melting in the case of alloy core would occur only a few milliseconds later than that of the aluminum portion of the oxide-type core.

The design-basis accident for the ORR which was specified by the ACRS[9] is a 100% core meltdown accompanied by a 100% release of noble gas and a 50% release of iodine inventories from the core. In the original analysis only minimal credit (a factor of three) was taken for the retention of iodine in the primary water. It is now known that this factor should be much higher (of the order of 100). Hence, although we do not believe that for any given situation the fission-product release would be significantly different in the case of oxide from that in the case of alloy, the consequences of the design-basis accident would certainly not exceed those already analyzed.

The operating conditions of the HFIR are considerably more rigorous than those of the ORR, as is illustrated in Table 2. This table also gives information on the NASA-converted 265-g elements which have already been used in the ORR as well as the estimated conditions for a 300-g element. It follows that the failure of an oxide element in the ORR is less likely than is the case in the HFIR.

Table 2. Characteristics of Various Fuel Elements

	ORR 240 g (Standard)	ORR 265 g (NASA Conv.)	ORR 300 g (Proposed)	HFIR Inner	HFIR Outer
Number of plates	19	18	19	171	369
^{235}U per plate (g)	12.63	14.72	15.79	15.20	18.34
Inner plate thickness (mils)	50	60	60	50	50
Outer plate thickness (mils)	65	66	65	-	-
Meat thickness (mils)	20	20	20	~20 avg	~18 avg
Meat length (in.)	23 5/8	23 5/8	23 5/8	20	20
Meat width (in.)	2.40	2.40	2.40	3.10	2.80
Channel width (mils)	116	113	116	50	50
Meat volume/plate (cm^3)	18.666	18.666	18.666	20.995	17.206
Loading (g/cm^3)	0.6766	0.7889	0.8459	0.7240	1.0711
Avg heat flux (Btu/ft^2hr)	1.65×10^5	1.65×10^5	1.65×10^5	7.7×10^5	
Max heat flux (Btu/ft^2hr)	7.2×10^5	$<7 \times 10^5$	$<7 \times 10^5$	2.5×10^6	
Max plate temperature (°F)	210	NA	NA	300	
Avg fission density (f/cm^3)	6.9×10^{20} [a]	8.9×10^{20} [b]	1.04×10^{21} [c]	5.64×10^{20}	6.08×10^{20} [d]
Peak fission density (f/cm^3)	1.2×10^{21} [e]	1.5×10^{21} [e]	1.7×10^{21} [e]	1.9×10^{21} [d]	

[a] Assumes 50% burnup.

[b] 55.5% burnup.

[c] Assumes 60% burnup.

[d] Based on 2300 MWD core life; total burnup 30.6% (65.6% outer, 34.4% inner).

[e] Calculated using ORR fuel code.

4. INCREASES IN FUEL WEIGHT

Several years ago the ORR fuel weight was increased from 200 to 240 g. This change resulted in a reduction of annual usage from about 150 elements to 100 elements. Recent experience using 265-g elements leads us to believe that an additional increase in weight could result in further fuel savings. Our estimates are shown in Table 3.

Table 3. Estimated Fuel Savings

Initial Fuel Weight	Annual Fuel Usage (Elements)
240 g	100
260 g	89
280 g	80
300 g	74

Before implementing an increase in fuel weight beyond that which has already been used, it is intended to mount a development program[10] to determine the feasibility of reducing the fuel element usage by such a weight increase. There are two aspects of this problem which require investigation. The first is purely operational and is simply that of determining what beginning fuel weight will result in the most economical operation and at the same time will provide a suitable reactor environment in terms of neutron flux distribution and fuel cycle to satisfy the needs of the experimenters. The second is to ensure that the restrictions on power density implied in Part B above can be satisfied.

Increases in fuel content up to the 40 wt % U_3O_8 (410 g ^{235}U per ORR element) already in use in the HFIR have been evaluated in connection with that reactor as noted above. This, together with the highly satisfactory performance of the HFIR fuel, leads to the conclusion that there are no previously unreviewed safety questions associated with an increase in fuel content up to the HFIR value.

It is unlikely that fuel loadings as high as 410 g will be required, and a value of 350 g ^{235}U (35.6 wt % U_3O_8) has been selected as a conservative upper limit. In this connection, it is worth noting that the Brookhaven High Flux Beam Reactor (HFBR) is currently using 351-g elements

(36.4 wt % U_3O_8) fabricated in a manner quite similar to that proposed for the ORR.

Nevertheless, the results of the development program will be evaluated and reviewed by the appropriate agencies prior to operation of the ORR at power with fuel containing greater amounts of ^{235}U than that previously used.

5. CONCLUSIONS

It may be concluded that there is no significant increase in the consequences of the accidents heretofore considered. We find no possibility that the proposed change can create an accident or malfunction of a different type than those previously evaluated. Moreover, the probability of a fuel failure is, if anything, reduced because of the greater strength and uniformity of the oxide element. Consequently, the change to oxide fuel does not present us with a previously unreviewed safety question.

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