MASTER

FUEL ELEMENT FLOW BLOCKAGE
IN THE
ENGINEERING TEST REACTOR

F. R. Keller

PHILLIPS
PETROLEUM
COMPANY

ATOMIC ENERGY DIVISION

NATIONAL REACTOR TESTING STATION
US ATOMIC ENERGY COMMISSION

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Appendices by

W. S. Little and J. H. Ronsick

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Atomic Energy Division

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A B S T R A C T

On December 12, 1961, the Engineering Test Reactor experienced fuel fission breaks which resulted from obstructions to the flow of cooling water through several fuel elements. A total of six fuel elements had breaks in from one to eight fuel plates.

This report discusses the event and describes damage to the fuel. The reactor operating conditions preceding the break are described, including events immediately before the manual scram. Results of investigations conducted after the scram, including effects upon the reactor as well as fuel element damage, are reported.

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I. SUMMARY

The Engineering Test Reactor experienced fission breaks in six fuel elements on December 12, 1961, as a result of obstructions to the flow of coolant in one quadrant of the core.

The incident was preceded by at least two indications of abnormalities in operating characteristics of the reactor. One was of relatively long duration, the second lasted less than a minute. A slow, but steady drop in flow of cooling water through the core (in the order of 100 gpm/min) was recorded over a 35 to 40 minute period and noise in Safety Level traces was discernible for about one minute before the first rupture took place.

It is known now that the first fresh fission products in the coolant stream were detected by the N-16 monitor, which showed an activity rise severe enough to cause an automatic power reduction. The reactor was brought back to power and operated at 90 Mw for five minutes before it was manually scrammed.

Examination of the core after the shutdown disclosed the presence of considerable amounts of foreign material blocking off several fuel elements in the northeast quadrant of the reactor. This material was later identified as Lucite from a sight box which had escaped detection during pre-startup inspections and checks.

Eighteen separate fuel plates, distributed among six fuel elements, were found to have melted in varying degrees. An estimated 12.4 grams of U-235, contained in 134 grams of alloy, were lost. Part of this material was recovered in cleanup operations but some remained in the system in the form of small bits and pieces.

Release of activity to the environment during, and immediately following, the break was minor. It is estimated that 6.4 curies of activity were released to the atmosphere, and 42 curies released to the leaching pond as liquid waste. Release of considerably higher amounts of activity, both to the atmosphere and to liquid waste facilities continues as a result of fission of the unrecovered U-235 remaining in the system. These releases amount to approximately 180 and 28 curies per day to atmosphere and liquid waste facilities, respectively. No over-exposures or severe Health Physics problems were encountered during or subsequent to the break. However, activity levels in the reactor coolant system remain considerably higher than those present before the ruptures, and background activities "plated out" in the reactor tank also are higher.

Hydraulic and thermal analyses of the event, based upon recorded data and current theory, indicate that a large part of the flow loss observed occurred in the six fuel elements found to have melted plates, and that the average flow in these elements was reduced to some 35% of normal. Under these conditions of flow it appears that alternate steam blanketing and refilling of channels ("chugging") was taking place, with fuel plate temperatures rising to the burnout point during the blanketing

cycle.

No evidence of a metal-water reaction was observed or found during or after the incident.

II. DISCUSSION

A. Operating Information

1. Conditions Before the Break. On December 12, 1961, the Engineering Test Reactor was being brought to power following a shutdown occasioned by a hardware failure in one of the loop experiments. The reactor had been refueled during this shutdown, and had reached a power level of 90 Mw (full power is 175 Mw) when a series of events took place which culminated in ruptures in one or more plates of six fuel elements.

The startup had been uneventful from criticality to 90 Mw. The only instrument difficulties or abnormalities observed during this period were:

a. Number 1 Safety Level was not responding properly to the power increase. With the instrument test block pulled, the reactor electronic instrument technician made a minor adjustment to preamp output and the instrument was returned to service. It should be noted that approved operating procedures allow startup with two of three levels operating; and the adjustments were made with the reactor at approximately 90 Mw.

b. The filter tape on the stack particulate monitor was not in motion. Gearing on the tape drive system had failed during the preceding cycle, and repair of the system was awaiting arrival of new gears. This condition meant that the detection head on the monitor was seeing integrated particulate activity rather than transient activity. A more pronounced instrument response would be expected under these circumstances.

c. One very small "blip" was noted on the N-16 monitor after the reactor had reached 90 Mw, but some 30 minutes before indications of real trouble developed. This monitor occasionally indicates very small short-cycle increases in N-16 activity which may be instrument noise, or which may be real. It is known that a hot particle passing the N-16 chamber will record as a blip of this nature. Therefore, this momentary rise caused no particular concern.

The reactor had been at a power level of 90 Mw for some 45 minutes when the first of several indications that something was amiss became apparent. The primary coolant flow recorder began showing a slow downward trend in flow of water. Approximately 35-40 minutes elapsed between the first indication of this trend and the time the reactor was scrammed. The flow loss during this period was about 4000 gpm. Thus the rate of loss of flow was 100 gpm/min. Usual recorder noise covers a range of 1000 gpm on the flow chart. These numbers are defined in such detail to point out that one cannot, with such an instrument, detect trends of the magnitude experienced within short periods of time. After-the-fact study of the chart shows that the trend was established within ten minutes, but this could be detected only with confirming extrapolation of the trend.

It is perhaps reasonable to expect that it could have been recognized within 15 minutes after it started. It was identified as a trend in about 20 minutes; however, the action taken was to check the instrument before accepting the drop in flow rate as real. This check was underway when the second indication of difficulty appeared.

The second indication was so short-lived, and of such a small magnitude, that it must be considered to have had doubtful usefulness to the reactor operator in determining that he was in trouble. However, its detectable presence on recorder charts makes it significant in retrospective analysis of the event. Specifically, all three Safety Level channels showed a short period of low-level noise immediately preceding the third indication of trouble.

The third, and by far the most pronounced indication of difficulty appeared as a spike on the N-16 monitor system of a magnitude sufficient to activate a reactor safety circuit. This action came as a rundown (motor-driven insertion of all withdrawn rods, reducing power to 1% of initial power in approximately 70 seconds). The rundown was cleared by the time reactor power had dropped to 70 Mw. At this point a decision was made that a second instrument was suspect. The instrument was separated from the power reduction circuitry (the N-16 monitor is considered to be a secondary power indicator, and it is not essential that it be in service while the reactor is in operation) and the reactor returned to 90 Mw. It is now apparent that this N-16 spike came as a result of the first release of fission products.

2. Conditions During the Break. The reactor remained at 90 Mw for five minutes after it had been returned to this power. During this interval the following abnormalities were observed:

a. Cooling water flow continued to decrease at a rate of 100 gpm/min.

b. All three Safety Level instruments exhibited mildly erratic behavior. The traces made by these instruments on recorder charts during stable operation are ordinarily very precise and look, on a strip chart, almost as if they had been drawn with a ruler. During this five minutes the trace was "noisy", in that the width of the line drawn increased some five fold.

c. The N-16 monitor trace was wildly erratic. Had this instrument not been removed from the power reduction circuitry, it would again have instituted an automatic power reduction.

d. With the reactor in servo, regulating rod action became highly erratic and it was extremely difficult for the console operator to keep it within limits.

e. Log N and Period traces showed no discernible abnormality.

f. Both Count Rate Meters displayed very erratic behavior.

g. Stack particulate and stack gas activity monitors displayed a steady rise during the five-minute period that the reactor was at 90 Mw. The rises were steeper than would be expected from the usual increases that accompany reactor power increases.

h. Some, but not all, area monitors indicated increases in background activity. Most pronounced indications were seen on the first floor north key way and north side of the reactor, and on the console floor at the south pipe tunnel.

i. The process water fission break monitor (Iodine Monitor) was inoperable at the time of the break because of a plugged feed line. It is to be noted, however, that this monitor probably would not have identified this break as such during the five-minute period because of the combination of its normal startup response, and its "built in" delay time.

j. All three Levels and both Count Rate Meters showed small, but definite indications of a power rise.

3. After the Fact Analysis. An analysis of the recorded data, the log books, and reports by reactor supervision reveals a clear and detailed picture of the event.

The first evidence of trouble to come, the indicated slow drop in water flow through the core, was difficult to evaluate in its early stages. It could have been false as a result of instrument drift; it could have been real, but not necessarily as a result of core flow blockage; or it could have been (as was subsequently established) a real flow drop resulting from core obstruction. The indicated change in this single variable could not, by itself, have been the basis for a conclusion that there was an obstruction to the flow of coolant through fuel elements in the reactor.

Immediately preceding the obvious and drastic N-16 monitor action there was a less apparent, but very definite instrument action which warned of abnormal behavior in the neutron populations seen by the Safety Level ion chambers. This action was in evidence for about one minute before it was interrupted and obscured by the power reduction which came from the N-16 instrument. During this short period of time, however, these instruments had to be seeing effects of void formation in the obstructed section of the core. A measure of the time it took to progress from first indication of void formation to rupture of the first fuel plate can be inferred if one accepts the premise that the N-16 monitor drove upscale because it saw fission products from the rupture.

The N-16 system has a built in delay time such that N-16 activity resulting from residence in the core is seen by the detector some five seconds later. Thus, approximately 55 seconds elapsed from the first indication of void formation to the first fuel plate rupture.

The void coefficient in this section of the ETR core is extremely low and negative. Early measurements in the ETR Critical Facility

resulted in a value of $-6 \times 10^{-8} \Delta k/k/cc$ for position N-6. A check of this value was made in the ETRC after the incident under conditions more nearly approximating those in the reactor. The coefficient was still negative, but larger by an order of magnitude (-6×10^{-7}).

It is believed that the voiding was in the nature of a "chugging" action. Both Safety Level charts and CRM charts tend to support this postulation in that fluctuations were recorded with discernible time intervals between spikes, although the relative effects of voiding and reg rod movement cannot be separated. The detailed thermal and hydraulic analysis of the incident (Appendix 2) also supports the chugging theory.

With intermittent voiding in one or more fuel elements in one quadrant of the reactor, and the negative void coefficient, a drop in fission rate occurred in the affected section. The decreased fast neutron leakage created two mechanisms which contributed to the apparent power rise in other sections of the core.

(1) A local, and relatively severe drop in neutron population took place in, and closely adjacent to, this quadrant.

(2) The effects of this drop were felt to a lesser extent all across the core, with a corresponding tendency for a general drop in neutron population.

No doubt both effects were being seen by the Number 1 Servo chamber in its location adjacent to the unstable core volume. The reaction of the servo system to these conditions would be to withdraw rod to restore reactor power to the setpoint neutron level seen by the chamber. A power rise in other sections of the core is a consequence of this response.

The Reactor Physics and Engineering Branch is planning a detailed analog study of the observed core behavior and will report their results at a later date.

After the power reduction initiated by the N-16 monitor the reactor was returned to 90 Mw. In retrospect it is, of course, obvious that this action should not have been taken. Nevertheless it was, and the reactor remained at this level for about five minutes. During these five minutes further indications of trouble became progressively worse, and all are readily explainable as consequences of either fission product release or boiling in the core.

4. Thermal and Hydraulic Aspects of the Incident. A detailed evaluation of the thermal and hydraulic effects encountered during the incident has been made, and is incorporated into this report as Appendix 2, Page 29.

5. Post Break Operations. Following the manual scram the reactor was degassed and the primary coolant started through the by-pass demineralizer. Flow through the demineralizer continued for several hours, followed by two 60,000 gallon flushes. Primary coolant system samples taken before, during, and after these operations showed steadily

decreasing activity level. When the reactor top was taken off no gaseous activity was encountered and the water in the vessel was at a very low activity level (less than 1 mr/hr from a 100 cc sample).

Inspection showed that a substantial number of fuel elements in the NE quadrant of the core were covered to some degree with a resin-like material. (See Figure 1). It was first thought that it was indeed resin from our process water demineralizer, but further examination, confirmed by chemical analysis, showed the material to be an acrylate resin. It was concluded that this resin could only have come from a Lucite sight glass, and that such a sight glass must have been left in the tank and escaped detection when the reactor was being prepared for startup.

Work was started to clean up the core and adjacent reflector area. All surfaces that could be reached were thoroughly vacuum cleaned with an underwater vacuum cleaner. Nineteen fuel elements in the affected quadrant were removed to the storage canal. A number of pieces of the sight glass were located and removed with finger tools. All control rod fuel sections were inspected for debris and each rod was checked for freedom of action in the guide tube. No evidence of any structural damage to core components could be found.

After two days of cleanup new fuel elements were inserted to replace those removed and preparations were made to start circulation of process water, following usual startup routines. Detailed destructive inspection of the fuel elements removed had not been made at this time and the full extent of the damage to fuel was not yet suspected.

After the top dome was installed a program of circulation of process water was started, since it was anticipated that some debris from the sight glass might still be in the system. After three hours of circulation the system was shut down, the dome removed, and the core reinspected. A relatively small number of particles, some of which were metallic, were observed and removed. Some of these metallic particles were saved and found to contain U-235 upon subsequent analysis.

The top dome was again installed, circulation started, and the reactor brought to criticality, following usual startup routines. Full power of 175 Mw was reached at 2:00 a.m. on December 15, 1961. A period of operation followed during which activity levels in process water, and releases of activity through the stack to the environment, were followed closely. Power levels varied during this period, and when it became apparent that general activity levels were continuing to rise the reactor was shut down at 1530 on December 18, 1961, to carry out further cleanup work. Details of the activity problems encountered during this period are presented in the Health Physics Aspects section of this report.

Following shutdown the reactor was degassed and flushed twice, and the top dome removed. Scrubbing operations were started on all tank surfaces and additional pieces of debris were removed from fuel and reflector areas. The cleanup during this period was extensive and included checks on all fuel elements, control rod fuel sections, and most core and reflector experiment pieces, since it was now known that the

system contained an appreciable amount of unclad uranium and fission products in the form of bits and pieces. On December 23, 1961, a four-hour circulation, followed by thorough flushing, was conducted. The core was cleaned and vacuumed one more time following this, the dome was put back on, and the reactor brought to criticality at 1954 hours.

Since this time, a considerable amount of effort has been expended during each reactor shutdown to search out and remove remaining particles. The system activity, though still higher than pre-break levels, is not seriously affecting operation of the reactor.

B. Fuel Damage

1. Preliminary Inspection. The full extent of fuel involvement was not known when the reactor was shut down on December 12, 1961. Preliminary inspection of the core revealed that several fuel elements, all in the northeast quadrant of the reactor, appeared to be either partially or completely blocked with the resin-like material described earlier. This section of the reactor is complicated with inpile tubes and braces, and is relatively difficult to inspect. Figure 1 is a picture shot of the area before any fuel or foreign material was disturbed. Figure 2 is a reactor core schematic which shows the position of the affected area in relation to the whole core. Fuel element numbers are shown on both figures to aid in identification.

All fuel which showed any indication of foreign material, together with two fuel elements far removed from the affected area, were removed from the core. These latter two elements were to be used as "blanks" in more detailed inspections in the storage canal. Nineteen suspect elements, plus the two check elements, were removed.

These twenty-one elements were run through a "stagnation" rupture test apparatus after having been visually inspected. This stagnation test consists of immersing the individual fuel element in a closed container of demineralized water for 20 minutes, sampling the water, and scanning the sample for I-132. Results of this scan are then compared with background and an estimate made of the probability of a fuel break. As a matter of interest, this test selected eight elements out of the twenty-one as being ruptured or possibly ruptured. All six elements found by hot cell examination to have been ruptured were included in these eight. Table I compares results from the three inspection methods used.

After the preliminary inspections were complete, nine elements were selected for further destructive testing in the MTR-ETR Hot Cell. Eight of these were suspect, the ninth (U-1550) was felt to have a very low probability of rupture but was to be examined as a test of the selection methods used. When U-1550 proved to be undamaged it was concluded that it was not necessary to destroy more than the eight suspect elements in the hot cell inspection program.

Six of the nine elements examined were found to have one or more melted fuel plates. A total of nineteen plates showed evidence of some degree of melting, and an estimated 12.36 grams of U-235 contained in

Fig 1 Section of Reactor Core
Immediately After Shutdown

Fig 2 Reactor Core Arrangement Schematic

134 grams of total metal was lost. No outside fuel plates were found damaged. Appendix 1 presents detailed photographic data on the ruptures.

TABLE I
COMPARISON OF INSPECTION RESULTS

<u>Fuel Element No.</u>	<u>Visual Inspection</u>	<u>Ruptured as Predicted by</u>	
		<u>Stagnation Method</u>	<u>Hot Cell Examination</u>
U-1550	No	Low Possibly	No
U-1558	Yes	Yes	Yes
U-1559	Yes	Yes	Yes
U-1560	Yes	Possibly	Yes
U-1564	Possible	No	No
U-1565	Yes	Yes	Yes
U-1567	Possible	Possible	Yes
U-1570	Yes	Yes	Yes
U-1573	Possible	No	No

C. Health Physics Aspects

1. Release of Activity. The fission break itself resulted in negligible exposure to personnel. It was not necessary to evacuate any operating area and there was no spread of contamination other than that contained within the ETR Primary Coolant System. There was an immediate, but very small, release of radioactivity to the environment through the ETR stack. This amounted to:

Particulate Activity (Fission Gas Daughters)	0.4 Curies
Fission Gas Activity	6.0 Curies

Increased amounts of activity continue to be released as a result of fissioning of the unclad U-235 remaining in the system after the fuel ruptures. These releases are through the stack or to the MTR-ETR retention basin-leaching bed system. Stack releases are summarized in Table II. Power generation data for the same periods are included because of the dependence of stack activity upon power generation.

It is evident that stack releases to date as a consequence of the fuel rupture have been in the order of ten times the levels in existence during the three-month period preceding the break.

Activity released to the retention basin-leaching bed system is not measured as such because of the joint use of these facilities by both MTR and ETR. A rough approximation of the amount can be inferred from the difference between total activity releases to the leaching ponds before the break and after the break. This amounts to 28 curies per day increase for the period between December 12, 1961, and March 18, 1962, and is a factor of three above "normal".

TABLE II

ETR STACK PARTICULATE AND GAS ACTIVITY RELEASES
DECEMBER 14, 1961, TO MARCH 18, 1962

<u>Curies of Activity Released</u>				
<u>Period</u>	<u>Power Produced</u> <u>(MwD.)</u>	<u>Particulate</u>	<u>Fission Gas</u>	<u>Total</u>
Dec. 14 to 31, 1961	2011	278.3	4917	5195.3
January, 1962	2006	237.4	4880	5117.4
February, 1962	2438	145.3	3339	3484.3
March 1 to 18, 1962	<u>2976</u>	<u>179.8</u>	<u>5342</u>	<u>5521.8</u>
Total	9431	840.8	18,478	19,318.8
<u>Sept. 1 to Nov. 30 *</u>	8270	75.2	1,511	1,586.2

* Listed for comparison

2. Primary Coolant Activity. Immediately after the fuel rupture reactor process water (primary coolant) samples were reading as high as 1 r/hr at contact. This activity died off at typical fresh fission product decay rates. Problems from fission products released during the rupture were not severe, since very little fuel burnup had occurred prior to the break. An estimated 42 curies of mixed fission products were released to the effluent system in reactor flushes immediately following the break.

The major activity problem stems from the presence in the system of significant amounts of unclad U-235. Particles of melted fuel alloy have been scattered through the system and cannot be 100% recovered in spite of intensive cleanup and decontamination efforts. As these particles pass through, or lodge in, high flux locations in the reactor core, release of fission products to primary coolant continues.

Figure 3 shows the activity trend of the process water from the time of the break through March 17, 1962. Also shown are trends in the activities released to the ETR stack. Power days have been adjusted to represent substantially full power levels, and shutdown periods are eliminated from the plot. In general, each abrupt drop in each of the

Fig 3 Activity Levels following ETR Fission Break

curves is indicative of a shutdown and coolant system flush and thorough core vacuuming. Such shutdowns took place on the 4th, 13th, 17th, and 28th power days.

In terms of predicting conditions to be expected in the future, the last 27 days at power are most significant. During this period the reactor ran at, or near, full power with no shutdowns. Although activities rose slightly during the nearly four weeks, all were within tolerable limits, and at much lower average levels than were encountered earlier. The rises in discharges to the stack late in the period were a consequence of a release from an inpile gas-cooled loop and are independent of the fuel rupture. No severe problems are anticipated from remaining U-235 in the system, and it is expected that over-all activity levels will be lower during the next operating period.

3. Deposited Activity. A major source of concern was the possibility that fission product activities would "plate out" at high equilibrium levels on internal reactor tank and experiment surfaces in contact with the primary coolant. If this indeed took place it could result in a buildup of long-lived activity which might eventually result in intolerable radiation levels for shutdown work on top of, and within, the reactor vessel. The ETR Health Physics group has set up a surveillance program to follow this trend. This program involves placing two sets each of stainless steel and aluminum coupons of known surface area in contact with primary coolant in the reactor vessel. Coupons are removed and counted on a 256 channel gamma spectrometer after each operating period. One set is returned to the reactor to track cumulative buildup, a second, fresh, set is inserted to trace "cycle" accumulation. Results to date from the program are summarized in Table III.

It is evident that there is no reason to expect a serious buildup of fission product contamination on reactor surfaces. Of the nuclides being followed, only Zr-95 is showing any significant increase. While this activity might possibly become a problem, it is felt that trend will not increase as the inventory of fissionable material in the system is reduced by burnup.

TABLE III

DISPOSITION OF SELECTED FISSION PRODUCTS FROM REACTOR COOLANT

c/m/m²

On Stainless Steel

Cycle	Days Operation	Ru-103		Zr-95		Ce-141		I-131		Ba-140-La-140	
		Cycle	Accum.	Cycle	Accum.	Cycle	Accum.	Cycle	Accum.	Cycle	Accum.
42D	9	3.33x10 ⁶	3.33x10 ⁶	3.04x10 ⁶	3.04x10 ⁶	2.05x10 ⁷	2.05x10 ⁷	3.69x10 ⁶	3.69x10 ⁶	1.19x10 ⁶	4.19x10 ⁶
43A	4	1.05x10 ⁶	*	1.53x10 ⁶	*	1.31x10 ⁷	*	1.5 x10 ⁶	*	1.3 x10 ⁶	*
43B	11	1.06x10 ⁶	3.39x10 ⁶	2.23x10 ⁶	4.3 x10 ⁶	1.4 x10 ⁷	1.74x10 ⁷	4.58x10 ⁶	3.25x10 ⁶	1.29x10 ⁶	8.3 x10 ⁵
44	27	1.95x10 ⁶	3.51x10 ⁶	2.39x10 ⁶	6.02x10 ⁶	1.72x10 ⁷	1.46x10 ⁷	1.52x10 ⁶	2.38x10 ⁶	1.07x10 ⁶	1.03x10 ⁵

On Aluminum

42D	9	1.3 x10 ⁷	1.3 x10 ⁷	6.95x10 ⁶	6.95x10 ⁶	2.1 x10 ⁷	2.1 x10 ⁷	5.2 x10 ⁶	5.2 x10 ⁶	Not Detectable on a gross spectrum - less than 10 ⁴	
43A	4	4.6 x10 ⁶	*	3.9 x10 ⁶	*	6.3 x10 ⁶	*	3.75x10 ⁶	*		
43B	11	4.66x10 ⁶	8.1 x10 ⁶	7.3 x10 ⁶	1.05x10 ⁷	2.45x10 ⁷	1.0 x10 ⁷	3.57x10 ⁶	3.57x10 ⁶		
44	27	8.76x10 ⁶	7.36x10 ⁶	7.95x10 ⁶	1.33x10 ⁷	6.36x10 ⁶	7.95x10 ⁶	2 x10 ⁶	3.33x10 ⁶		

III. ACTIONS PLANNED OR EXECUTED AS A RESULT OF THE BREAK

The incident would not have occurred if the sight box had not been left in the tank, and from analysis of the incident it is evident that fuel damage could have been limited, but probably not eliminated completely, if:

1. A different set of conclusions had been reached on the basis of the information available before the return to power.

2. A full scram, initiated by the high activity in the primary coolant, had been a part of the reactor protective circuitry.

All sight boxes have been replaced with a new design which includes red side walls, double-buoyancy chambers, and a chain from the box to a fixed point outside the reactor tank. However, these precautions do not preclude 100% the possibility that flow through a fuel element, or fuel elements, will be blocked again. Other measures being taken to provide further protection include:

1. A new fission break monitor, to be installed in addition to present Iodine and N-16 instruments, is under design. This instrument is to be specific in its function -- to shut down the reactor when it sees excessive amounts of activity in the cooling water leaving the core, and to initiate the shutdown in a matter of seconds. It must be relatively insensitive to N-16 or other so-called "normal" activities, and it must not be so sensitive as to respond to minor changes.

2. A review of the reliability of all major reactor instrumentation is underway. This data will be used, with other information, to conduct a statistical analysis to determine the probability of coincidental failures, or malfunctions, of two or more reactor instruments. It is hoped that this analysis will result in data which will provide the reactor operator with a powerful tool to assist him in determining the difference between instrument trouble and real trouble.

3. A study is being made to determine the feasibility of installing a second flow instrument in the reactor control room which will indicate and record full scale over a relatively narrow flow range. Such an instrument would be highly damped but would show quickly and surely relatively small flow trends.

APPENDIX 1. HOT CELL EXAMINATION (J. H. Ronsick)

Prior to transfer to the Hot Cell end boxes were sawed off the suspect elements in the ETR canal. The first two elements inspected in the Hot Cell (U-1550 and U-1565) were disassembled using the following procedure:

The ends of each element were cast in "Cerro-Bend" to facilitate holding and to minimize fuel plate distortion during subsequent cutting operations. Both side plates were slabbed off using a milling machine, and the individual fuel plates then separated by melting out the "Cerro-Bend" with hot water.

It was found, after completing work on the first two elements, that radiation levels would permit use of a band saw to remove side plates from the remaining elements. This permitted quicker disassembly and significant savings in hot cell time.

The results of hot cell examinations, including the photographic evidence of plate melting found, are presented in the following section. The photographs showing end blockages of individual elements should be interpreted with caution, since removal of end boxes in the ETR canal must have dislodged some of the material originally present.

Element U-1550. The end view (Figure HC-1) of this element shows all flow channels to be open, although one small particle was found lodged between plates 3 and 4 upon disassembly. Hot spots on these two plates which resulted from this particle are shown in Figure HC-2. No other evidence of damage to this element was found.

Fig HC-1, End View of
Element U-1550

Fig HC-2, Hot Spots Between
Plates 3 and 4

Element U-1558. The end view of this element (Figure HC-3) shows extensive blockage to fuel channels. It was located in position K-5 in the core and was observed to have been almost fully blanketed with resin. This element cannot be seen in the photograph of the core (Figure 1) since an inpile tube obstructs the line of sight. In spite of the apparent severe blockage only two fuel plates were found damaged. Plates 5 and 6 had fused together over an extensive area. (Figures HC-4 and HC-5).

Fig HC-3, End View of
Element U-1558

Fig HC-4, Plates 5 and 6 fused
Together

Fig HC-5, Plates 5 and 6 fused
Together

Element U-1559. The end view of this element (Figure HC-6) shows partial or complete blockage in fourteen of the eighteen flow channels. It was located in position L-5 in the core and part of it is visible in the core photograph, showing correspondingly severe blockage. Eight plates of this element were melted, with damage ranging from one small hole to extensive fused areas. Plate 3 (Figure HC-7) had melted and fused to plate 4 (Figure HC-8). Plate 5 (Figure HC-9) had one small melted area. Plates 6 through 9 had fused together. Figures HC-10 and HC-11 show plates 6 and 9, respectively. One more plate, number 10, had a melted area as is shown in Figure HC-12. Metallic pieces appearing in many of these pictures fell out of the element while it was being dis-assembled.

Fig HC-6, Flow Channel Blockage in Element U-1559

Fig HC-7, Plate 3

Fig HC-8, Plate 4

Fig HC-9, Plate 5

Fig HC-10, Plate 6

Fig HC-11, Plate 9

Fig HC-12, Plate 10

Element U-1560. This element showed significant blockage (Figure HC-13). It was in position M-5 and reference to Figure 1 shows it to have been severely blocked. However, only one plate, number 6, was found melted (Figure HC-14). A large metal blob (Figure HC-15) fell out of this element during side plate removal.

Element U-1564. No obstructions or defects were found in U-1564. Its position in the reactor (J-6) cannot be seen in the core photograph, but examination in place indicated that there was little probability of damage.

Fig HC-13, Blockage of
Element U-1560

Fig HC-14, Plate 6

Fig HC-15, Metal Blob
From Plate 6

Element U-1565. The end view of this element (Figure HC-16) shows severe blockage in twelve channels and lesser blockage in others. It can be clearly seen in the core photograph (position N-6) as almost totally blocked off. Plates 3, 8, and 9 have melted, with the latter two fused together. Figure HC-17 is an enlarged view of the fused portion of plate 3, and shows that as melting occurred the molten material flowed out and against the adjacent plate and solidified. No visible damage was caused to the adjacent plate. Figure HC-18 shows one side of plate 3, and plate 8. Figure HC-19 shows the other side of plate 3, and plate 9.

Fig HC-16, End View Element U-1565

Fig HC-17, Fused Portion of Plate 3

Fig HC-18, Side of Plate 3, and 8

Fig HC-19, Side of Plate 3, and 9

Element U-1567. The end view of this element (Figure HC-20) shows relatively little blockage. However, the core photograph shows substantial obstruction (the element is in position N-7) and it is felt that much of the blocking material may have shaken loose or been removed when the end box was cut off. Significant melting occurred in two plates, plate 2 (Figure HC-21) and Plate 4 (Figure HC-22).

Fig HC-20, End of
Element U-1567

Fig HC-21, Melting in
Plate 2

Fig HC-22, Melting in
Plate 4

Element U-1570. The end view of this element (Figure HC-23) shows substantial blockage to flow channels, and the element appears almost completely blocked in the core photograph (position N-8). Plates 8, 9, and 10 were found to have melted areas. (Figures HC-24, HC-25, and HC-26).

Fig HC-23, End View of
Element U-1570

Fig HC-24, Plate 8

Fig HC-25, Plate 9

Fig HC-26, Plate 10

Element U-1573. This element was found to be clear and unobstructed, although the core photograph indicates considerable amounts of material on it (position N-9).

The results of the hot cell examination program are summarized in Table HC-I below. Data on reactor positions are included.

TABLE HC-I

RESULTS OF THE HOT CELL EXAMINATION PROGRAM

<u>Element Number</u>	<u>Position In Reactor</u>	<u>No. of Melted Plates</u>	<u>Wt. Loss Grams</u>	<u>U-235 Loss Grams</u>
1550	J-9	0	0	0
1558	K-5	2	24	2.25
1559	L-5	8	54	1.95
1560	M-5	1	7	.65
1564	J-6	0	0	0
1565	N-6	3	34	3.13
1567	N-7	2	5	.46
1570	N-8	3	10	.92
1573	N-9	0	0	0

The weight loss determinations leading to the values tabulated above were made by weighing two or more intact fuel plates from each element and comparing their weights with those showing evidence of melting. The differences in fuel plate weights so obtained were assumed to represent actual loss of metal to the reactor. A further assumption was made that the metal lost was 9.2% U-235 by weight. This assumption was derived from knowledge that fuel alloy is 23% by weight U-235 and estimating that fuel alloy represented 40% of the metal lost.

APPENDIX 2. HYDRAULIC AND THERMAL ANALYSIS

(W. S. Little)

INTRODUCTION

Figure T-1 is a core diagram showing the fuel element number and the per cent of total core power generated in each element for the Cycle 42B loading. Noted on this diagram are the elements which were covered with debris and which were inspected in the canal for channel blockage. Also noted are the elements which contained melted plates.

Figure T-2 shows graphically the plates which melted and the length of the melted area for each plate of each element. Throughout this report, plate No. 1 is the southernmost fuel plate of each element. It is assumed that the plates which melted were in the hottest area of each element which is the southern half of the element. This is shown in detail in Figures T-3 (a), (b) and (c) which present detailed thermal neutron flux distributions for position L-5, and in Figures T-4 (a), (b) and (c) which present the detailed thermal neutron flux distributions for position N-7. The thermal neutron flux distribution in position L-5 was considered typical for the elements in K-5, L-5 and M-5, and was used in the heat transfer calculations pertaining to the fuel elements in these positions. Also, the flux distribution for position N-7 was used in the heat transfer calculations pertaining to the fuel elements in N-6, N-7 and N-8.

All of the fuel elements in this loading were new and had not been in the reactor prior to Cycle 42B startup. Prior to and during this incident these elements operated at 90 Mw for approximately one hour. A review of the Phillips Petroleum Company's and the vendor's inspection sheets for the six melted fuel elements reveals that they were within the required specifications and of a good quality. In the calculations performed in this analysis the actual channel dimensions were used as obtained from the inspection sheets. The ETR fuel assembly contains 400 grams of U-235 and approximately 2.26 grams of natural boron distributed equally from plate to plate. The assembly has variable coolant channel spacing as shown in Figure T-7. The normal flow through an ETR element is approximately 550 gpm at a core pressure drop of 43 psi. The normal inlet water temperature is 110° F and inlet pressure is 200 psig.

Hydraulic Analysis

Table T-I presents the inspection results after the removal of loose material from the top of those elements which were covered with debris. This shows that the channel blockage was significantly greater in the six elements which contained melted plates. The amount of flow blockage does not necessarily correlate with the number of melted plates in any one element and apparently the amount of loose material on top of an element significantly affected the flow through the element. Because of the loose material on top of the elements it is difficult to make an estimate of the flow through the fuel elements. The fact that the six fuel elements containing the melted plates were the only elements which contained a significant amount of lucite within the coolant channels indicates that a large part of the 4000 gpm primary coolant flow reduction which occurred

during this incident took place within these elements.

Inspection of the core immediately after the incident revealed that the blockage occurred mainly in 17 fuel elements with the majority of the debris located on the north and east sides of the "99" experimental facility. A 4000 gpm reduction would give an average flow reduction per element of 235 gpm. From the inspection of the fuel assemblies it appears reasonable to estimate that the flow reduction through the six melted elements was at least 50% greater than the average, giving a flow reduction of 352 gpm for each of these six elements. This indicates that the average flow through the six elements was approximately 36% of normal. Judging from the element inspections there was a considerable variation in flow area restriction from element to element.

Hydraulic calculations have been made to determine the average amount of flow blockage necessary for a given reduction in flow. A summary of these calculations is presented in Figure T-5. These calculations were made assuming that all of the flow restrictions occurred at the channel entrance and that the restrictions did not proceed a significant distance from the channel. The calculations indicate that an 84% flow area restriction would reduce the total coolant flow through a fuel element by 64%. From the appearance of these fuel elements, considering the solid material inside the channels as well as the loose material on top of the elements, this estimate appears to be low. The coolant velocity would be approximately 11.5 ft/sec in a channel whose flow area restriction was 84%.

For an element whose flow is 36% of normal with an average flow area restriction of 84%, it is probable that the restriction would vary from channel to channel causing excessive lateral pressure differentials across the fuel plates which could result in fuel plate buckling. Calculations made to determine this effect are summarized in Figure T-6 and indicate that the fuel plates would be susceptible to buckling unless the flow area restriction was fairly constant from channel to channel. Inspection of the fuel plates from the six elements containing melted plates revealed only one case of plate deformation which could be attributed to excessive lateral pressure differential. This case appears in fuel element U-1565 in which Figure T-8 shows that fuel plate No. 8 is deformed significantly. Plates 8 and 9 melted and the way in which plate 8 deformed shows that the flow in channel 8 was less than that in channel 7. Figure HC-16 reveals that the flow blockage in channel 7 may be less than that in channel 8, which would be expected in order for plate No. 8 to deform as it did.

Significant variations in flow area restriction from channel to channel are shown in Figures HC-3, HC-6, HC-13, HC-16, HC-20 and HC-23 in some areas of each of the six fuel assemblies containing melted plates. Calculations indicate that these variations could cause fuel plate buckling. An example of this is fuel element U-1565 as shown in Figure HC-16 in which the flow area restriction is 72% in channel 10, 23% in channel 11, 74% in channel 12 and 31% in channel 13. The calculated lateral pressure differential which would exist across plate 13, separating channels 12 and 13, can be derived from Figure T-6. At the top of the fuel plate the lateral ΔP would be approximately 15 psi while at the bottom of the fuel plate the ΔP would be approximately 9 psi in the opposite direction.

Excessive deformation of plate 13 would be expected at the temperatures at which this plate operated and with these calculated lateral pressure differentials. The fact that no noticeable deformation occurred in plate 13 and in other plates under similar conditions proves that the differences in flow area blockage from channel to channel was not as severe as the presence of the solid material in the channels would indicate, and that the loose material on top of the elements must have contributed significantly to the flow reductions.

The following conclusions can be drawn from the hydraulic analysis:

1. A large part of the total flow reduction which occurred in the ETR core as a result of the sight glass shattering and covering the core with debris occurred in the six fuel elements which contained melted fuel plates.
2. An estimate of the resulting average flow rate through the six elements is 36% of the normal flow rate of 550 gpm per element.
3. Out of a total of 114 fuel plates in the six fuel elements containing melted plates, only one plate showed any deformation which could be attributed to lateral pressure differentials caused by an unbalance of flow from channel to channel. This indicates that the flow restriction from channel to channel was fairly uniform and that possibly there was little difference in the hydraulic conditions which existed from channel to channel in an element.

Heat Transfer Analysis

A summary of the heat transfer calculations is presented in Table T-II for the six elements containing melted plates. These values were calculated using the uncertainty factors listed in Table T-III. $V_{b.o.}$ is the velocity at which the Bernath ⁽¹⁾ burnout heat transfer correlation predicts that burnout would occur at the point of maximum heat flux. V_{sat} is the coolant velocity at which the exit coolant temperature reaches saturation. This would be accompanied by a reduction in coolant flow due to increased pressure drop resulting in subsequent fuel plate melting. Case I of Table T-II presents the calculated results for the individual channels for each element ignoring the conditions in adjacent channels. Case II presents the calculated results for two channels adjacent to a channel which is completely plugged off.

The calculations indicate that there would be film boiling in the core before the burnout of fuel plates would occur. This is confirmed by the behavior of the reactor safety levels which gave an indication that boiling was taking place for about one minute before there was any indication of high coolant activity resulting from the release of fission products. It should be pointed out that $V_{b.o.}$ and V_{sat} are very nearly equal and the accuracy of the burnout heat flux correlation would not give one the freedom to say that V_{sat} occurred before $V_{b.o.}$ unless the behavior of the reactor instrumentation supported this.

The Jens and Lottes (2) correlation indicates that nucleate boiling would begin in these elements considerably before the occurrence of V_{sat} or $V_{b.o.}$. In channel No. 3 of fuel element U-1559 it is calculated that nucleate boiling would begin at a flow rate of approximately 5.5 ft/sec, at a hot spot fuel plate temperature of approximately 413° F, as compared with 1.6 and 2.7 for $V_{b.o.}$ and V_{sat} respectively. It is possible that the initiation of nucleate boiling could significantly change the hydrodynamics in the coolant channels causing a drastic reduction in coolant flow causing burnout to occur prematurely; however, this is not supported by the behavior of the reactor instrumentation, and the author does not think that nucleate boiling would be detected by the reactor instrumentation. A survey of the nuclear field in December, 1959, by V. A. Walker (3) revealed that there had been no known instances in which nucleate boiling had been detected in water cooled reactors. The author knows of no developments in the last two years that would indicate that nucleate boiling would be detected with the existing ETR instrumentation.

The behavior of the reactor instrumentation indicates that some of the coolant channels in the six melted fuel elements were "chugging". By chugging it is meant that there was a cyclic filling of the channels with steam expelling the water and a subsequent refilling of the channels with water. Calculations, however, cause one to question whether this was actually taking place. In channel No. 6 of fuel element U-1570, which is the hottest channel in which melting occurred of the six melted elements, a maximum heat addition to the channel of 2.3 BTU/sec/in² is calculated. This value was obtained by assuming that all of the heat generated at the hot spot in both fuel plates forming the channel is transferred into the channel coolant. To vaporize enough water at its saturation temperature to fill one half of the coolant channel volume, it would be necessary for the coolant to remain stagnant at the hot spot for approximately 0.5 seconds with an infinite heat transfer coefficient between the water and the fuel plate surface. If the adjacent fuel plates were perfectly insulated at the hot spot for approximately 0.9 seconds, calculations indicate that the plates would melt. From these figures it appears that if chugging occurred the frequency would be approximately 0.5 to 2 cycles per second with the fuel plate temperature beginning to rise at the initiation of chugging and reaching the melting point soon after the chugging began.

If chugging did not occur and nucleate boiling cannot be detected, then what caused the erratic behavior of the reactor safety levels? Two theories can be thought of which would cause a coolant channel to fluctuate into and out of the boiling regime with the heat transfer rate remaining constant throughout the incident. One theory would be for the lucite to build up at the channel entrance restricting the flow and causing boiling to begin, but with the restriction partially breaking up and being washed out of the channel due to the pressure drop across the restriction increasing as the restriction increases. It does not appear likely that this would be fast enough to cause the erratic behavior of the safety levels, nor is there any evidence that it actually happened. The second theory might be that as melting occurred, heat generating material was removed from the channel, reducing the coolant temperature and causing boiling to cease until the flow area was further restricted again at the

channel entrance. However, it appears from the hot cell inspection that the molten fuel alloy flowed down the coolant channel and solidified in a cooler section. This action would increase the possibility of boiling since another flow restriction would be added to the channel and the heat flux would increase on the opposite surfaces of the fuel plates between which the molten metal solidified. The information from the hot cell inspection presented in Table HC-I indicates that the U-235 weight loss would not be sufficient to support this second theory. It is therefore concluded that chugging is still the best explanation for the erratic behavior of the safety levels.

One fact which supports the chugging theory is the evidence that out of 19 melted fuel plates, melting within three inches of the bottom of the fuel plate occurred in only three fuel plates. Only one of these was melted below the three-inch mark. It is theorized that in chugging the steam is produced in the center of a channel in the areas of highest flux and the cyclic expulsion of steam and sucking in of coolant kept the lower sections of the fuel plates sufficiently cool enough to prevent melting.

It was observed that in no instance did an outside fuel plate melt. In fuel elements which were in the "five" row of the reactor core (see Figure T-1) the outside fuel plate was not next to another fuel element and the outside channels could suffer a greater flow reduction than the inside channels before burnout could occur. However, in the "N" column of the core the outside fuel plates were next to another fuel plate of the adjacent element and still melting did not occur. Due to the tolerance allowed between fuel elements it is possible that the channel between fuel elements is slightly larger than any internal channel. This leads one to conclude that the particle size of the debris was small enough to pass through the channels between elements without severely restricting the flow area.

It should be mentioned that there is no evidence that an explosive metal-water reaction occurred when the fuel plates melted.

It is concluded that:

1. Fuel plate melting occurred as a result of the channel exit coolant temperature reaching saturation resulting in "auto-catalytic" vapor binding and therefore burnout. This was caused by the gradual flow area blockage throughout the incident.
2. The erratic behavior of the reactor instrumentation was evidently caused by "chugging".

Fig T-1 Cycle 42B Fuel Loading
and Power Generation

Fig T-2 Location of Melted Areas
Within Fuel Elements

PPCo.-B-4098

Fig T-3a Horizontal Thermal Neutron Flux Distribution in Position L-5

Fig T-4a Horizontal Thermal Neutron Flux Distribution in Position N-7

Fig T-3b Vertical Thermal Neutron Flux Distribution in Position L-5

Fig T-4b Vertical Thermal Neutron Flux Distribution in Position N-7

Fig T-3c Horizontal Thermal Neutron Flux Distribution in Channel 18 in Position L-5

Fig T-4c Horizontal Thermal Neutron Flux Distribution in Channel 5 in Position N-7

Fig T-5 Coolant Channel Flow Vs. Channel ΔP
For Different Restrictions at the Entrance of a Coolant Channel

Fig T-6 Pressure Drop Through an ETR Fuel Element Channel
For Various Flow Area Restrictions With Total Flow
Through the Element 36% of Normal

P.P. CO. - C-2344

Fig T-7 ETR Fuel Element Channel Spacing

Fig T-8 Fuel Element U-1565 Bottom View

TABLE T-I
FUEL ELEMENT INSPECTION AFTER REMOVAL OF LOOSE MATERIAL
FROM THE TOP OF THE ELEMENTS

<u>Element No.</u>	<u>Reactor Position</u>	<u>Comments</u>
U-1547	I-7	Flow area clear except for small pieces of lucite 1/8 - 1/4" wide.
U-1548	I-8	Flow area clear except for small pieces of lucite 1/8 - 1/4" wide.
U-1549	J-8	Flow area clear except for small pieces of lucite 1/8 - 1/4" wide.
U-1550	J-9	Flow area clear except for small pieces of lucite 1/8 - 1/4" wide.
U-1551	L-9	Flow area clear except for small pieces of lucite 1/8 - 1/4" wide.
U-1552	K-10	Flow area clear except for small pieces of lucite 1/8 - 1/4" wide.
U-1553	M-10	Flow area clear except for small pieces of lucite 1/8 - 1/4" wide.
U-1554	N-10	Flow area clear except for small pieces of lucite 1/8 - 1/4" wide.
U-1557	I-5	Flow area clear except for small pieces of lucite 1/8 - 1/4" wide.
U-1558	K-5	Approximately 1/2 of total flow area blocked.
U-1559	L-5	Approximately 3/5 of total flow area blocked.
U-1560	M-5	Approximately 2/3 of total flow area blocked.
U-1563	I-6	Flow area clear except for small pieces of lucite 1/8 - 1/4" wide.
U-1564	J-6	Large No. of small pieces 1/8 - 1/4" wide. 1/4 of flow area in one channel blocked.
U-1565	N-6	Approximately 3/5 of total flow area blocked.
U-1567	N-7	Approximately 1/3 to 1/2 of flow area blocked in several channels.
U-1570	N-8	Approximately 1/3 of total flow area blocked.
U-1573	N-9	Approximately 1/4 of flow area blocked in several channels.
U-1594	N-11	Flow area clear except for small pieces of lucite 1/8 - 1/4" wide.

TABLE T-II

SUMMARY OF HEAT TRANSFER CALCULATIONS
FOR FUEL ELEMENTS CONTAINING MELTED PLATES

		CASE I			CASE II		
Fuel Element No.	Channel No.	(Q/A) Max. BTU/hr ft ²	V _{b.o.} ⁽²⁾ ft/sec	V _{sat} ⁽³⁾ ft/sec	(Q/A) Max. BTU/hr ft ²	V _{b.o.} ft/sec	V _{sat} ft/sec
U-1558	1 (1)	4.99 x 10 ⁵	2.5	3.1			
	4	3.61 x 10 ⁵	2.0	2.5			
	5	3.48 x 10 ⁵		2.5			
	6	3.35 x 10 ⁵		2.4			
U-1559	1 (1)	5.09 x 10 ⁵	2.8	3.0			
	2	4.34 x 10 ⁵		2.7	7.96 x 10 ⁵	4.55	4.07
	3	3.98 x 10 ⁵	1.6	2.7	This channel plugged		
	4	3.73 x 10 ⁵		2.7	7.46 x 10 ⁵	4.55	4.08
	5	3.54 x 10 ⁵		2.7			
	6	3.42 x 10 ⁵		2.4			
	7	3.32 x 10 ⁵		2.3			
	8	3.25 x 10 ⁵		2.4			
	9	3.21 x 10 ⁵		2.4			
	10	3.21 x 10 ⁵		2.4			
U-1560	1 (1)	4.42 x 10 ⁵	2.2	2.6			
	5	3.08 x 10 ⁵	1.6	2.2			
	6	2.96 x 10 ⁵		2.1			
U-1565	1 (1)	3.77 x 10 ⁵	1.6	2.2			
	2	3.77 x 10 ⁵		2.3			
	3	3.77 x 10 ⁵	1.9	2.3			
	7	3.7 x 10 ⁵		2.4			
	8	3.65 x 10 ⁵		2.4			
	9	3.6 x 10 ⁵		2.4			
U-1567	1 (1)	5.54 x 10 ⁵	3.0	3.1			
	2	5.54 x 10 ⁵		3.2			
	3	5.54 x 10 ⁵		3.2			
	4	5.53 x 10 ⁵	3.2	3.5			
U-1570	1 (1)	6.13 x 10 ⁵	3.5	3.6			
	7	6.01 x 10 ⁵	3.6	3.9			
	8	5.94 x 10 ⁵		3.9			
	9	5.86 x 10 ⁵		3.7			
	10	5.81 x 10 ⁵		3.7			

(1) Channel 1 data is presented since the max. heat flux in an element occurs in this channel. Melting did not occur in any of these channels.

(2) V_{b.o.} is the velocity at which burnout is predicted by the Bernath correlation.

(3) V_{sat} is the velocity at which the avg. water temp. leaving the channel reaches the saturation temp.

TABLE T-III

UNCERTAINTY FACTORS USED IN CALCULATIONS

Hot Channel Factor

Reactor power measurement error	1.05
% core power per element error	1.06
Error in flow measurement	<u>1.05</u>

$$F_{\Delta T_b} = 1.17$$

Heat Flux Factors

Reactor power measurement error	1.05
% core power per element error	1.06
Local fuel content variation	1.05
Variation of fuel core alloy area	<u>1.08</u>

$$F_{Q/A} = 1.26$$

Hot Spot Factors

Variation in velocity within channel	1.18
Correlation equation (modified Colburn)	<u>1.25</u>

$$F_{\Delta T_f} = 1.48$$

Additive temperature measurement error +3°F

Burnout was calculated as occurring at a heat flux equivalent to 0.6 of the value predicted by the Bernhath burnout correlation.

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3. V. A. Walker, "Nucleate Boiling Detection Survey", Phillips Internal Report, December 14, 1959.