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TECHNICAL BASES
FOR
FFTF DRIVER FUEL INSTRUMENTATION

OCTOBER 1968

AEC RESEARCH & DEVELOPMENT REPORT

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This document supersedes an earlier report, BNWL-555, with the same title.
Technical bases for instrumentation requirements to protect the core of the Fast Flux Test Reactor are evaluated in detail for driver fuel. Power excursions, loss of flow and fuel failure are characterized to establish sensitivity and response time requirements. Surveillance and reliability requirements are also considered. Preliminary safety criteria relative to core instrumentation are formulated and brief rationale presented. A comprehensive assessment of present instrument technology is presented and used as a basis for establishing instrument development goals.
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TECHNICAL BASES FOR
FFTFT DRIVER FUEL INSTRUMENTATION

I. OBJECTIVES

The Fast Flux Test Reactor is being designed to accommodate a broad range of fuel and material irradiation experiments. An initial evaluation was carried out to identify the factors involved in instrumenting driver and test fuel to accomplish the design goals. These considerations were taken into account in arriving at a choice between various reactor concepts. Driver fuel which provides the nuclear and thermal environment for irradiation tests must be monitored, not only to assure the required testing conditions but to prevent the fuel from exceeding thermal ratings which may place the reactor and experiments in jeopardy. Since the thermal ratings of the driver fuel involve core design considerations, the requirements for core instrumentation have been determined by evaluating first the characteristics of the design which might bring about excessive temperatures. A second consideration is the instrumentation available to prevent these occurrences from progressing far enough to endanger the core integrity.

Technical bases for the driver fuel instrumentation have been divided into two categories. The first is classed as reactor safety, and the second as surveillance.

A. Reactor Safety

Instrumentation upon which the safe operation of the reactor is dependent falls in this category. Rapid indication of impending loss or actual loss of driver fuel integrity is required and sensors may actuate the reactor safety circuit. Occurrences are characterized as follows:
Coolant Flow Choking or Blockage
Fuel Cladding Failure
Accidental Reactivity Insertion (Overpower)

Each of these occurrences gives rise to changes in the local environment. Sensors must be capable of detecting the change above the established background level rapidly enough to initiate reactor shutdown. Therefore, one of the objectives of this study is to establish the time associated with the change in the local environment to be sensed resulting from the above occurrences.

B. Surveillance

Included in the surveillance category is the instrumentation that allows more efficient operation. Although not directly depended upon for reactor safety, surveillance instrumentation can provide trend information to give advance warning of developing problems. Specifically, a periodic indication of the core environment is required that will permit the reactor to operate closer to safeguards limits and eliminate unnecessary conservatism. The conditions to be detected are:

- Temperature Distribution
- Flow Distribution
- Neutron Flux Distribution

Bases for requirements of surveillance instrumentation are in general less demanding than for reactor safety instrumentation. Thus, the objective of the present study in this regard is simply to estimate uncertainties in the above distributions and the changes that occur over relatively long time intervals.

Finally, an overall objective is an assessment of the capability of various types of instruments to meet indicated requirements. Included in this assessment are the response time, sensitivity, resolution and lifetime characteristics based on present technology. Since on-going
development will result in improvement in the present state of the technology, an attempt is made to use the goals of these programs to estimate what improvements in the above characteristics may result by the time the instruments are needed.

The FTR core consists of multiple pin subassemblies within individual flow ducts through which the primary sodium coolant is circulated. Coolant enters the core from below and flows upward through the active zone. Coolant exits from the subassembly duct at a point above the fission gas plenum, which is an extension of the fuel pin above the active zone. The dispersed core geometry of the reference concept permits full instrumentation in each subassembly flow channel without interference or interaction from adjacent channels.

A power density near 0.5 MW/liter for the driver fuel requires high heat removal capability and leaves little temperature margin below incipient fuel melting for reduction of coolant flow or for overpower accidents. Therefore, a primary requirement for driver fuel instrumentation is that it provide sufficient advance indication of either coolant flow reduction or overpower such that corrective action can be instituted. A second requirement in the FTR is that the instrumentation provide experimenters with a measurement of the environment in which tests reside. The present evaluation deals only with the operational and safety requirements. It is, of course, evident that operational and safety objectives can coincide with experimental objectives as well since sensing of the environment is involved in either case. Instrumentation requirements relating to closed and open-loop experiments were evaluated in a follow-on study utilizing some of the analyses presented herein. (2)
II. SUMMARY

Detailed analyses have been carried out for power excursions, loss of flow and release of fission products to the sodium coolant. Power excursions were investigated for a range of reactivity inputs, trip settings, time delays, rod accelerations and reactivity coefficients. Loss of flow was considered from the standpoint of both flow coastdown and flow blockage. Fission product release was analyzed to obtain estimates of fission product concentration in the coolant from a break in the fuel cladding. Conclusions from these analyses are as follows.

A. Power Excursions

In a power excursion the first limit reached is that of fuel melting. For a 2$/sec reactivity input ramp the core outlet temperature increases only 27°F when the fuel begins to melt. It is concluded that it is not possible to detect ramps of this magnitude or larger by coolant temperature monitoring in time to prevent fuel melting. Thus, reliance must be placed on the nuclear instrumentation for this class of accident. Limiting safety system parameters required to prevent fuel melting were found for ramp reactivity additions of 2$, 4$, and 6$/sec. Combinations of safety rod acceleration, instrument trip setting and time delay of 3 g, 1.25 P 0 and 100 msec respectively will prevent melting for ramps as high as 6$/sec. However, a delay time of 200 msec cannot be tolerated for this maximum ramp.

B. Loss of Flow

Flow coastdown simulating loss of all pumps was considered for three detection methods: loss of power, low flow, and high coolant ΔT. For initiation of scram 0.75 sec after loss of power or for scram at 60% of full flow, coolant and clad temperatures are within safe levels. For scram initiation at 150% coolant ΔT sensed at a point above the fission gas plenum relative to the core inlet, the peak coolant temperature reached was 1384°F. The time to reach boiling temperature is sufficiently
long that a flowmeter with a time constant of the order of 1 sec would be acceptable. If \( \Delta \)-temperature detection is to be considered, the time constant for thermocouple sensing can also be in the range of 1 to 2 sec depending upon the desired trip setting. In any case, it appears that for this class of accident, either temperature or flow detection instrumentation will suffice.

Flow blockage is extremely difficult to analyze because of the uncertainty in type and severity of the occurrence. Since this area remains largely speculative, the analyses were carried out for severe blockage conditions without regard for credibility. Blockages occurring outside the fuel region, leading to a uniform reduction of flow, were investigated and found to result in the sodium coolant temperature just reaching the boiling point at the core outlet for 63% and 52% reductions in flow for two different fuel pin designs, respectively. It was found that prevention of coolant boiling for an instantaneous reduction in flow of 80% is not possible, even for no delay in sensing this condition. It is possible, however, to prevent boiling at 70% reduction providing reasonable time delay exists in the flow sensor. This type of occurrence requires a very fast sensor and reactor safety system. Flow blockage of less than 50% is much more probable and either flow or channel outlet temperature sensors will provide an indication of a potential worsening condition. The salient value of individual channel flow sensing is in detecting a blocked channel in advance of going to power since it is shown that coolant thermocouples above the fuel bundle will not be able to detect the blockage.

A more credible occurrence within the loss of flow category is sub-assembly subchannel blocking or choking. This can result from fuel swelling, fuel pin distortion, lodging of foreign objects or broken hardware. Blockages of one and six subchannels were analyzed. For a single subchannel blockage, it was found that it would be impossible to detect the local temperature disturbance at the end of the fuel bundle, even if a conservative amount of coolant mixing is assumed. It was also found that the clad temperature can reach magnitudes sufficient to induce failure. For a
six-subchannel blockage, it is possible to detect the local temperature
disturbance at the end of the fuel bundle only if a coolant thermocouple
is located immediately above the blocked region. Excessive clad tempera­
tures can also be obtained if the axial extent of the blockage is greater
than 0.050 in. long.

The reduction in subassembly flow due to 24 blocked subchannels
assuming a constant pressure drop across the subassembly causes only a 3% 
increase in the coolant $\Delta T$ across the fuel bundle ($11^\circ F$ for a $350^\circ F$ $\Delta T$).
Therefore this degree of blockage or equivalent fuel swelling may not be
detected by subassembly flow or temperature measurement and some other
method of detection must be utilized. Two methods that show promise are
thermocouple signal fluctuation analysis and acoustic sensors. However,
both of these techniques are experimental at this time and cannot be set
as firm requirements for the first core.

C. Failed Element Detection and Location

Requirements for the detection of fuel failure within a fuel assembly
stem largely from the need to rapidly detect loss of fuel cladding integrity.
The basis for this need is principally the large uncertainty in progression
of a local failure into one of major consequences. Fuel pin cladding is
considered to be the first barrier in preventing fission products from
escaping the reactor but more importantly it also defines the coolant
passages within the core. Thus a local defect can distort the temperature
field, either through choking or gas blanketing, causing overheating that
results in further failure and so on in an autocatalytic manner. Considered
in its most pessimistic light, such a failure, if undetected, might progress
rapidly to a significant core meltdown.

Consideration of systems which are intended to detect fuel pin faults
need not be restricted to those which monitor directly for material which
has leaked or burst through a flaw in the cladding. Failure may be sig­
naled by characteristic acoustic noise or by failure-consequent thermal
phenomena or coolant flow reduction. But here we have examined separately
the possibility of monitoring for released material, especially gases, because of the unique way such an indication constitutes proof of failure. Thus the Failed Element Detection and Location (FEDAL) system gives special emphasis to the possibilities of monitoring for active fission products and implanted tags for which the cladding normally provides gas-tight containment.

The activity of the total inventory of contained, noble gases in an FTR fuel pin at goal exposure is expected to be about 7500 curies and even soon after the beginning of a fuel cycle (~1500 MWd/T) nearly half that much. Some 80% of this material may be trapped at interstitial sites in the pin, depending upon operating conditions. If one-tenth of the remainder were released upon clad failure an activity of 150 curies would be released to the coolant. It is concluded that the FEDAL system should be capable of reliably detecting a fuel pin failure resulting in a noble gas release of at least this order.

Release of delayed neutron precursors from fuel pin defects has been analyzed. The concentration of active precursors in the fuel pin gas plenum is expected to be low because of long diffusion times for migration from the fuel structure into this region. Possible release of precursors directly to the FTR coolant was compared with data on a bar pin experiment in EBR-II using an analytical model in which near-surface recoil release of bromine and iodine fission fragments was assumed. Because of uncertainties in thermalization and counting efficiency in the EBR-II experiment, it is concluded that postulates of precursor release based on actual count rate in that work are reliable only to within about a factor of ten. Extrapolation of this model and information relative to FTR fuel is further complicated by uncertainties in the retention of precursors in the fuel and between fuel and clad in the FTR and their speed of migration to possible defects in cladding. Based on available information, an assessment of the fraction swept out of an FTR fuel pin defect during instantaneous depressurization yields a source strength between $10^5$ and $10^8$ neutrons/sec released to the coolant in a burst. Therefore, basing the FEDAL system on delayed neutron detection would require discrimination of sources of this order.
Release of implanted tag isotopes affords still another method for detection of pin failures. Insufficient information has been developed at this time to relate the release of such isotopes to fuel failures. However, the methods are similar to those used for noble gas and delayed neutron estimates given above.

Location requirements for the PEDAL system are not considered to be a matter affecting reactor safety, providing action is taken to remove the failed fuel. The principal bases for locating the affected fuel assembly are plant availability and costs. Two approaches may be considered to the location of failed fuel within the core. First, provision for locating the assembly can be coincident with the detecting function. Alternatively, location can take place at some later time, provided that discriminating evidence of the failure is retained. Either approach is acceptable within the requirement that the location time not be chargeable against plant availability. Thus, the requirement that such location be accomplished within refueling access time (cooldown plus preparation) will assure that a reasonable latitude exists in design of the locating method.

It is evident for purposes of location that the maximum discrimination required is a single fuel assembly; i.e., it is not required to identify failed pins within an assembly. Methods proposed wherein discrimination can only be accomplished between groups of assemblies are judged to be less desirable, but must be considered on tradeoff if a more highly reliable, less complex system, results. Single assembly discrimination is required for purposes of conceptual design.

D. Surveillance

This class of instrumentation is deemed essential because of the continually changing loading requirements of the FFTF, the need to identify anomalous behavior and trends which may precede more hazardous circumstances, and to remove unnecessary conservatism in operation that would prevail if analysis alone were relied upon for margins to limits. Sensors are not required to have high speed to perform a surveillance function, although
sensors may be serving a safety function simultaneously, in which case the latter requirements are overriding.

It is estimated that periodic surveillance of temperature, flow and neutron flux distributions over the core will result in an improved power density capability of between 5 and 10%. Simultaneous comparison of these distributions with predictions is recommended as an effective diagnostic of anomalous behavior. This information will be especially useful following a loading change and before the reactor reaches full power.

It is concluded that the minimum requirement for surveillance is sub-assembly outlet temperature distribution, subassembly $\Delta T$ per MW and neutron flux distribution. Greater assurance yet can be achieved from flow distribution over the core.

E. Reliability Requirements

Reliability requirements were compared for (1) failure of the instrument function in a single core assembly and (2) failure of the instrument function in two adjacent assemblies. A reactor unavailability of 0.002 is allocated for each case. In the first case the reactor is assumed shut down and the failure in that channel repaired. For the single failure criterion, it was found that the one-year instrument reliability must be approximately 0.997 in order for the overall reliability to be 0.731.

The adjacent tube failure criterion implies that shutdown will occur for failure of any two adjacent assembly instrument functions. For this case a one-year reliability of 0.96 for the instrumentation is required to produce an overall reliability of 0.833.

Application of the above results for each criterion was extended to in-channel thermocouple circuits. For a failure rate of 0.1 per year, four installed circuits are required for the single channel criterion and three circuits satisfy the adjacent channel criterion. Provision for four circuits is recommended at this time.
No analysis of potential failure indicated the practical applicability of the adjacent tube failure criteria inasmuch as signal strength is greatly reduced in adjacent channels. The mathematical formalism for the adjacent tube failure criterion was prepared in the event that future studies may identify situations where its usefulness is based in technical fact.

F. Conceptual Design

To assist in the visualization of the application of requirements to design, two drawings have been included in an appendix to this report. The first of these, SK-3-14323, shows the reactor instrumentation system in schematic form, together with interfaces with other systems. A separate group of instruments for monitoring the reactor vessel is also shown.

The second drawing, SK-3-14194, shows a conceptual layout of the instrumented fuel assembly. This design incorporates an instrument package that is removable, with the instrument section of the assembly above the fuel section. Incorporated in the instrument package are temperature, flow and failed fuel sensors.

It is not intended to depict the disposition of alternate instrument systems which relate best to each of the variety of reactor concepts which continue to merit the consideration of FFTF designers, but these drawings can provide an indication of those factors which influence instrumentation system integration. Separate design descriptions, as they become available, furnish the most recent and specific views of component layouts.
### G. SUMMARY TABULATION OF REQUIREMENTS for Driver Fuel Instrumentation

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<th>Useful Sensor Range</th>
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<th>Instrument Capability, Development Goals</th>
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<td>Limiting condition--incipient fuel melting</td>
<td>Shutdown to 300 percent design power.</td>
<td>Ion Chamber</td>
<td>Adequacy uncertain for high gamma background for startup and intermediate ranges.</td>
<td>Determine adequacy of chambers in high gamma background and determine shield design.</td>
<td>Response time &lt;100 msec from 10 to 1000 MW.</td>
<td></td>
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<td>2. Neutron Flux Period/Rate</td>
<td>Approx. 0.01 to 7 sec.</td>
<td>Ion Chamber</td>
<td>Same as A.1 above.</td>
<td>Same as A.1 above.</td>
<td>Response time not determined; to be determined in a future study.</td>
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<td>3. Coolant Temperature</td>
<td>Ambient to Sodium Boiling Temperature</td>
<td>Thermocouple</td>
<td>Not applicable for fast transients.</td>
<td>Not applicable for fast transients.</td>
<td>Response time 1-2 sec for slow transients.</td>
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<td>Limiting condition--coolant boiling</td>
<td>0-125% Rated Flow</td>
<td>Ion Chamber</td>
<td>Adequate for smaller pipes and lower temperatures.</td>
<td>Prove accuracy, linearity, for large pipes and 1200 °F.</td>
<td>Response less than one second.</td>
<td></td>
</tr>
<tr>
<td>2. Pump Speed</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Limiting condition--coolant boiling</td>
<td>10-100% Maximum Speed</td>
<td>Flowmeter</td>
<td>Adequate with proper design.</td>
<td>None needed.</td>
<td>One-second response to speed drop.</td>
<td></td>
</tr>
<tr>
<td>3. Pump Electrical Power</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>4. Subassembly Coolant Temperature</td>
<td>Ambient to Sodium Boiling</td>
<td>T/C</td>
<td>Adequate for temperature range but flux exposure effect largely unknown.</td>
<td>Reliability and accuracy, T/C materials and insulation, repeatability, in fast flux.</td>
<td>1/2 sec from initiation of decay of field on relay.</td>
<td></td>
</tr>
<tr>
<td>5. Subassembly Flow</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Limiting condition--coolant boiling</td>
<td>0-125% Rated Flow</td>
<td>Flowmeter</td>
<td>Radiation effects largely unknown.</td>
<td>Fast response, ±10% accuracy, resolution to ±%, temperature capability to 1200 °F.</td>
<td>Response less than one second.</td>
<td></td>
</tr>
<tr>
<td>6. Pump Discharge Pressure</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Limiting condition--coolant boiling</td>
<td>0-125% Rated Pressure</td>
<td>Flowmeter</td>
<td>Adequate with proper design.</td>
<td>None needed.</td>
<td>One-second response to pressure change.</td>
<td></td>
</tr>
</tbody>
</table>

**NOTE:** Response time defined as time to sense 63% of expected value resulting from a step change.
5. SUMMARY TABULATION OF REQUIREMENTS (continued)

<table>
<thead>
<tr>
<th>Conditions to be Detected</th>
<th>Parameter Sensed</th>
<th>Useful Sensor Range</th>
<th>Sensor Type</th>
<th>Present Instrument Capability</th>
<th>Instrument Capability, Development Goals</th>
<th>Initial Core Instrumentation Requirements</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Fluctuation</td>
<td>Boiling</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>2. Subassembly</td>
<td>0-125°F Rated</td>
<td>Subassembly Flowmeter</td>
<td>Same as B.1.5 preceding.</td>
<td>Same as B.1.5</td>
<td>Detection of 60% flow minimum. Response &lt;1 sec.</td>
</tr>
<tr>
<td></td>
<td>Flow Rate</td>
<td>Flow</td>
<td></td>
<td></td>
<td></td>
<td>Backup detection method. Detect incipient local boiling.</td>
</tr>
<tr>
<td></td>
<td>3. Boiling Noise</td>
<td>Pre-to-Post</td>
<td>Acoustic</td>
<td>Experimental</td>
<td>Long-range LMFBR Program.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Boiling</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Fluctuation</td>
<td>Boiling</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>2. Boiling Noise</td>
<td>Pre-to-Post</td>
<td>Acoustic</td>
<td>Experimental</td>
<td>Long-range LMFBR Program.</td>
<td></td>
</tr>
<tr>
<td></td>
<td></td>
<td>Boiling</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>C. Cladding Failure</td>
<td>1. Noble Gas</td>
<td>150 curies</td>
<td>Gas chromo., charged wire detector, or gross gamma.</td>
<td>Feasibility established. Equipment and system improvements needed.</td>
<td>Fission gas separation and sampling method improvement.</td>
<td></td>
</tr>
<tr>
<td></td>
<td>Fission Products</td>
<td>Burst</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>2. Delayed Neutrons</td>
<td>10^5-10^8 n/sec</td>
<td>Neutron Counters</td>
<td>Same as C.1 above.</td>
<td>Improvement and optimization of all system components.</td>
<td>Same as C.1 above.</td>
</tr>
<tr>
<td></td>
<td>Burst</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>3. Implanted Tags</td>
<td>Not Determined</td>
<td>Isotopes of Xenon</td>
<td>Experimental</td>
<td>LMFBR Program</td>
<td>Backup method.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>II. PROCESS SURVEILLANCE</td>
<td>A. Temperature</td>
<td>Subassembly</td>
<td>Ambient to</td>
<td>Subassembly Thermocouple</td>
<td>Adequate for temperature range, but unknown flux effect.</td>
<td>Same as B.1.4</td>
</tr>
<tr>
<td></td>
<td>Distribution</td>
<td>Coolant Temperature</td>
<td>Boiling</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>2. Subassembly</td>
<td>0-125°F Rated</td>
<td>PM or Eddy Current Subassembly Flowmeter</td>
<td>Same as B.1.5 above.</td>
<td>Fast response, ±10% accuracy, at 1200 °F.</td>
<td>Response set by Safety (B.1.5) Reproducibility to within 15% (calibration correction permissible).</td>
</tr>
<tr>
<td></td>
<td>Coolant</td>
<td>Flow</td>
<td></td>
<td></td>
<td></td>
<td>Flux profile from selected points locations not defined.</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>3. Neutron Flux</td>
<td>1 x 10^12 n/cm^2-sec</td>
<td>Ion chambers adequate.</td>
<td>Improved activation and transport. (FPPF development program deferred.)</td>
<td></td>
<td></td>
</tr>
<tr>
<td></td>
<td>Distribution</td>
<td></td>
<td>(thermal)</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
III. ANALYSES

A. Power Excursions

The rate of reactivity insertion must be limited to values such that the resulting power excursion can be terminated by the safety system. This design reactivity insertion rate should be as high as practical to avoid undue limitations on the facility in either the design or flexibility of operation. On the other hand, high insertion rates require more sophisticated designs of the control and safety systems to limit the time delays and to accelerate the control elements at greater rates upon release.

The physical characteristics of the fuel pin assumed in the analysis are summarized in Table I below:

<table>
<thead>
<tr>
<th>TABLE I</th>
</tr>
</thead>
<tbody>
<tr>
<td>DESCRIPTION OF HOTTEST FUEL PIN</td>
</tr>
</tbody>
</table>

| Pin length, inches       | 36  |
| Pin OD, inches           | 0.250|
| Pin pitch to diameter ratio | 1.12 |
| Clad thickness (stainless steel type 316), inches     | 0.016|
| Fuel pellet OD, inches   | 0.212|
| Original fuel density, percent theoretical density    | 93  |
| Sintered fuel density, percent theoretical density     | 98  |
| Maximum linear heating rate, KW/ft                     | 12.1 |
| Axial power peaking factor                                | 1.24 |
| Coolant inlet temperature, °F                            | 500  |
| Coolant temperature rise across core, °F                | 414  |
| Fuel-to-clad gap coefficient, Btu/hr-ft²-°F             | 1500 |
| Sodium film coefficient, Btu/hr-ft²-°F                  | 30,000|
For the above fuel pin, the fuel reaches incipient melting at 25% overpower. The coolant temperature rise of 414°F corresponds to a combined radial power peaking factor and engineering factor of 1.38 applied to a 300°F average coolant temperature rise across the core. An unorificed core is assumed.

It is assumed that during operation, the initial mixed-oxide pellet expands uniformly out to the clad resulting in a fuel density of 88% of theoretical. The fuel above 3272°F (1800°C) under normal operating conditions is then assumed to be sintered to 98% of the theoretical density, thus giving a varying axial void. This internal fuel relocation was predicted by the SINTR computer code.\(^{(3)}\)

For the transient analysis, the fuel pin was divided into 18 2-inch-long axial segments. Each segment consisted of seven internal (volumetric) nodes, an outer fuel surface node, an internal clad node, inner and outer clad surface nodes, and a coolant node. All of the power was assumed to be generated in the fuel, and axial conduction within the fuel, clad, and coolant was neglected. Figure 1 is a diagram of the hottest fuel pin with the steady-state temperatures at normal operating power.

The NUTIGER computer program\(^{(4)}\) was used for the transient analysis. NUTIGER is an expansion of the three-dimensional heat transfer code TIGER V to include nuclear feedbacks due to Doppler, coolant density change, fuel expansion, and fuel bowing. It solves the zero-dimensional, multigroup (up to six delayed neutron groups) nuclear kinetics equations after each time step and supplies TIGER with the new value of power. Changes in temperature are fed back to NUTIGER as changes in reactivity, thus causing subsequent changes in power. Transients can be initiated by time-dependent reactivity insertions or by changes in the coolant flow rate or inlet temperature. For these analyses, the reactivity insertion consisted of the input ramp and, after a given time delay, the reactivity worth due to scram of the safety system. The feedbacks used were those due to the Doppler effect and the sodium density change.
**FIGURE 1. Fuel Geometry and Temperature Distribution for Hottest Fuel Pin**

- **Temperature, °F**
  - Core
  - Fuel Inner: 1274°F
  - Fuel Outer: 970°F
  - Surface Clad: 920°F
  - Clad Coolant: 907°F
  - Outlet: 907°F

- **Sintering Interface**
  - 1800 °C (3272 °F)

- **Hottest Fuel Pin**
  - Maximum Heating Rate: 15.1 kW/ft
  - Coolant ΔT: 414 °F
  - Fuel-Clad Gap Coefficient: 1500 Btu/ hr-ft°F
  - Film Coefficient: 30,000 Btu/ hr-ft°F

- **P/A Axial**
  - 1.24

- **Radial Distribution**
  - 18 - 3124
  - 17 - 3469
  - 16 - 3725
  - 15 - 3934
  - 14 - 4085
  - 13 - 4197
  - 12 - 4278
  - 11 - 4327
  - 10 - 4345
  - 9 - 4331
  - 8 - 4285
  - 7 - 4207
  - 6 - 4098
  - 5 - 3954
  - 4 - 3748
  - 3 - 3495
  - 2 - 3159
  - 1 - 2675

- **Void**: Radius from Center of Fuel Pin, in.
Use of the hottest fuel pin with no orificing is quite conservative, and it is possible that a moderate amount of fuel melting in this pin would correspond to no fuel melting in the core. Figure 2 compares the maximum fuel temperatures for the average and hottest fuel pin for ramp rates of 2$/sec, 4$/sec, and 6$/sec. Although fuel melting occurs in the hottest fuel pin for the 4$/sec ramp, the maximum fuel temperature for the average fuel pin peaks out at almost 600°F below the melting point.

For power transients, fuel melting is reached before coolant boiling or clad melting. This is illustrated in Figure 3 for a ramp reactivity insertion of 2$/sec, which is at the lower end of the credible range of power excursions anticipated. When the hottest part of the fuel reaches the melting point, the temperature of the coolant at the core outlet has increased by only 27°F. Due to the very high sodium film coefficient compared to the fuel-to-clad gap coefficient, the clad temperatures also rise only slightly. Assuming that a coolant thermocouple must be located at the end of the fission gas plenum above the core, it can be concluded that it is impossible to detect reactivity insertion rates of 2$/sec or greater by means of coolant temperature monitoring before significant melting in the hottest fuel pin has occurred.

Table II shows the various power excursions initiated from normal operating power. The parameters varied were:

- Reactivity ramp rate: 2$/-, 4$/-, and 6$/sec
- Nuclear instrument trip setting: 1.05, 1.15, and 1.25
- Time delay from instrument detection to release of the rods: 50, 100, and 200 msec.
- Rod acceleration: 1/2, 1, and 3 g.
- Doppler coefficient \(\left(\frac{\Delta k}{\Delta T}\right)\): -0.002, -0.003, -0.004
- Sodium temperature coefficient \(\left(\frac{\Delta k}{\Delta T}\right)\): 2.0 \(\times\) 10\(^{-6}\) and -4.0 \(\times\) 10\(^{-6}\)/°C.
FIGURE 2. Fuel Temperatures in Hottest and Average Fuel Pins for Power Excursions
FIGURE 3. Temperatures for Ram Reactivity Insertion of 28/sec
<table>
<thead>
<tr>
<th>Case</th>
<th>Ramp Rate</th>
<th>Core Location</th>
<th>Trip Setting</th>
<th>Time Delay msec.</th>
<th>Rod Acceleration, G</th>
<th>Doppler Coefficient</th>
<th>Sodium Temperature Coefficient, °C^-1</th>
<th>Fuel Melting</th>
</tr>
</thead>
<tbody>
<tr>
<td>1A</td>
<td>2</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>100</td>
<td>1/2</td>
<td>-0.004</td>
<td>-4.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>1B</td>
<td>2</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>100</td>
<td>1/2</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>1C</td>
<td>2</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>100</td>
<td>1/2</td>
<td>-0.004</td>
<td>0</td>
<td>No</td>
</tr>
<tr>
<td>1D</td>
<td>2</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>100</td>
<td>1/2</td>
<td>-0.003</td>
<td>-2.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>1E</td>
<td>2</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>100</td>
<td>1/2</td>
<td>-0.002</td>
<td>-2.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>1F</td>
<td>2</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>200</td>
<td>1/2</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>1G</td>
<td>2</td>
<td>Average Pin</td>
<td>1.25</td>
<td>100</td>
<td>1/2</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>1H</td>
<td>2</td>
<td>Hottest Pin</td>
<td>No Scram</td>
<td>100</td>
<td></td>
<td></td>
<td>-2.0 x 10^-6</td>
<td>Yes</td>
</tr>
<tr>
<td>2A</td>
<td>4</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>100</td>
<td>1/2</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>Yes</td>
</tr>
<tr>
<td>2B</td>
<td>4</td>
<td>Hottest Pin</td>
<td>1.15</td>
<td>100</td>
<td>1/2</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>Yes</td>
</tr>
<tr>
<td>2C</td>
<td>4</td>
<td>Hottest Pin</td>
<td>1.05</td>
<td>100</td>
<td>1/2</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>2D</td>
<td>4</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>100</td>
<td>1/2</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>2E</td>
<td>4</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>50</td>
<td>1/2</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>2F</td>
<td>4</td>
<td>Hottest Pin</td>
<td>1.05</td>
<td>200</td>
<td>3</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>2G</td>
<td>4</td>
<td>Average Pin</td>
<td>1.25</td>
<td>100</td>
<td>1/2</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>3A</td>
<td>6</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>100</td>
<td>1/2</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>Yes</td>
</tr>
<tr>
<td>3B</td>
<td>6</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>100</td>
<td>1</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>Yes</td>
</tr>
<tr>
<td>3C</td>
<td>6</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>100</td>
<td>3</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>3D</td>
<td>6</td>
<td>Hottest Pin</td>
<td>1.05</td>
<td>100</td>
<td>1</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>Yes</td>
</tr>
<tr>
<td>3E</td>
<td>6</td>
<td>Hottest Pin</td>
<td>1.25</td>
<td>50</td>
<td>1</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>No</td>
</tr>
<tr>
<td>3F</td>
<td>6</td>
<td>Hottest Pin</td>
<td>1.05</td>
<td>200</td>
<td>3</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>Yes</td>
</tr>
<tr>
<td>3G</td>
<td>6</td>
<td>Average Pin</td>
<td>1.25</td>
<td>100</td>
<td>1/2</td>
<td>-0.004</td>
<td>-2.0 x 10^-6</td>
<td>Yes</td>
</tr>
</tbody>
</table>
The initial position of the tips of the poison rods was assumed to be 10 cm above the top of the core. The total worth of the safety system was assumed to be 20$.

The maximum fuel temperature as a function of time is plotted in Figure 4 for the various cases in Table II. Since the effect of varying the Doppler coefficient and sodium temperature coefficient was found to be very small for the 2$/sec ramps, these two parameters were not varied for the higher ramp rates. It can be concluded that for ramp reactivity insertions of 2$/sec or less, a relatively slow safety system (1.25 trip setting, 200-msec time delay, 1/2-g rod acceleration) will suffice to prevent fuel melting.

However, the same safety system will not prevent fuel melting for a 4$/sec ramp. To avoid melting, it is necessary to either:

- Lower the instrument trip setting from 1.25 to 1.05, or
- Reduce the time delay from 100 to 50 msec, or
- Increase the rod acceleration from 1/2 to 1 g.

A 200-msec time delay can be tolerated only if a low trip setting (1.05) and a high rod acceleration (3 g) are used.

For the 6$/sec ramps, the only two cases which produced maximum fuel temperatures below the melting point involve a delay time of 50 msec or a rod acceleration of 3 g. A delay time of 200 msec cannot be tolerated even for a trip setting of 1.05 and a 3-g rod acceleration.

Other scram transient analyses have been carried out(5) and the results correlated in terms of a "fuel overheating factor" which is computed by:

\[ F_{OH} = \frac{(T_f - T_{c,z})}{(T_f - T_{c})} \]
FIGURE 4. Maximum Fuel Temperatures for Power Excursions
where
\[
\hat{T}_f = \text{peak fuel center temperature during the transient at axial location } z
\]
\[
\hat{T}_c = \text{coolant temperature at axial location } z \text{ when } T_f = \hat{T}_f
\]
\[
\overline{T}_f = \text{fuel center temperature at steady-state full power at axial location } z
\]
\[
\overline{T}_c = \text{coolant temperature at axial location } z \text{ when } T_f = \overline{T}_f
\]

When the fuel undergoes melting, the amount of energy absorbed in melting must be converted into an equivalent temperature increase. Fortunately, it is found the fuel overheating factor is relatively constant in the axial direction and can be calculated within the accuracy of the results by using the fuel nodes which do not melt.

Figure 5 shows the fuel overheating factor as a function of reactivity insertion rate for five different combinations of system delay times and rod accelerations. The differences between these analyses and the previous analyses of Table II and Figure 4 are: (a) the power at the beginning of the transient was assumed to be 110% of design operating power to account for any operational fluctuations before initiation of the transient, and (b) the total worth of the safety system was assumed to be 10% to account for the possibility of failure of some safety rods to scram. It is seen that for a $4/sec ramp, fuel melting is avoided with either a 1-g rod acceleration and 50-msec time delay or a 3-g rod acceleration and 100-msec time delay. The results are replotted in Figures 6 and 7 showing the effects of rod acceleration and time delay.

The fuel overheating factor is important only if it can be related to fuel damage severity thresholds. These thresholds have been tentatively set at (a) incipient fuel melting, (b) centerline fuel completely molten, and (c) 50% of the cross-sectional area of the fuel pin at or above the fuel melting point and corresponding to negligible, moderate, and gross fuel
Power at Start of Transient = 110% Design Power
Scram Signal = 125% Design Power
Rod Worth = 10$
Doppler Constant $\frac{\Delta k}{\Delta T} = -0.003$
Sodium Temp Coef. = $-2 \times 10^{-6} / ^\circ C$

FIGURE 5. Maximum Transient Fuel Overheating Factor
 FIGURE 6. Effect of Safety Rod Acceleration
FIGURE 7. Effect of Time Delay to Rod Release
damage, respectively. Some type of corrective action before resumption of reactor operation should be established for each damage threshold.

B. Loss-of-Coolant Accidents

1. Flow Coastdown

A flow coastdown is a gradual flow decay such as that caused by loss of electrical power to the pumps. In a flow coastdown, coolant boiling is reached before fuel or clad melting. Figure 8 shows the flow rate, power, and maximum coolant temperature for the hottest fuel pin for a 3-sec eddy current clutch time constant. For the hottest fuel pin the coolant temperature at the core outlet reaches the normal boiling point (1620°F) at approximately 6 seconds if no scram occurs.

If the loss of flow is detected at 3/4 sec after the loss of electrical power, corresponding to a trip setting of 45 cycles for the 60-cycle power supply, and if the scram rods start to move after a 200-msec time delay at an acceleration of 1/2 g, the coolant temperature rises to a maximum of slightly over 100°F over normal before falling due to the decreased power level caused by scram. For the same time delay and scram rod acceleration, detection by a 60% trip setting on a flowmeter results in coolant temperatures peaking at slightly under 1200°F. If it is assumed that the loss of flow must be detected by a coolant thermocouple which is separated from the core by four feet of fission gas plenum, then the additional transport time (assuming constant hexagonal duct shape and heat transfer with the plenum) results in a significant temperature lag at the thermocouple location. For a very conservative trip setting of 150% on the coolant ΔT, and an increased time delay of 500 msec, scram is initiated in sufficient time to limit the maximum coolant temperature to below 1400°F. Therefore, both coolant flow and temperature measurements in the subassembly are capable of detecting a flow coastdown and scrambling the reactor to prevent coolant boiling.

Simulation studies of the reactor and heat transport system (6) have been carried out for loss of electrical power transients under the following conditions:
FIGURE 8. Temperature Response to Flow Coastdown for Various Scram Initiations
Three heat transport loops
400 MW\textsubscript{t} initial reactor power
600°F reactor coolant inlet temperature and 300°F average core ΔT
Reactor scram 0.75 sec after power failure
1-g rod acceleration
Total worth of safety system 15$ or 6$
Emergency pump speed controller setpoint is 100 rpm
2-sec eddy current clutch time constant

Figure 9 shows the hot channel outlet temperature for a 6-, 12-, and 18-ft elevation difference between the core and intermediate heat exchanger using the 15$ safety system. It appears that pony motors or natural circulation will provide adequate core cooling in loss of electrical power transients to prevent hot channel coolant boiling.

2. Instantaneous Flow Reduction

Occurrences which can cause a uniform reduction of flow in the fuel bundle include a blockage at the inlet to the fuel subassembly and a pipe break in the primary loop. Since it is difficult to assess the amount of flow reduction and the rate at which the flow decreases, various levels of instantaneous flow reductions were arbitrarily assumed.

Figure 10 shows that a 63% flow reduction causes the coolant outlet temperature for the hottest fuel pin to reach 1620°F, the boiling point of sodium at 1 atm. The results for a 90% flow reduction show that it is impossible to prevent coolant boiling even for instantaneous detection with a flowmeter, allowing for a 100-msec time delay and a 2-g rod acceleration. However, boiling is avoided for a 70% flow reduction with instantaneous flowmeter detection, 100-msec time delay, and a 1-g rod acceleration.

To determine the effect of rate of flow reduction on the detection and prevention of boiling, a 90% flow reduction was assumed to occur at 100-, 60-, and 40% per second. For a 90% flowmeter trip setting, 100-msec time
FIGURE 9. Hot Channel Core Outlet Temperature from Simulation Studies for Loss of Electrical Power

3/4 sec Delay, 1n (150) Reactor Scram
Flow Decay to Natural Circulation with \( hZ = 6, 12, \) and 18 ft
FIGURE 10. Temperature Response of Coolant to Flow Reductions: 15.1 kW/ft Hottest Fuel Pin
delay, and 2-g rod acceleration, the rate of flow reduction must be between 40 and 60% per second to avoid boiling.

Analysis of instantaneous flow reductions was also carried out for a slightly different hottest fuel pin in a later stage of the conceptual design. Some of the parameters which are different from Table I are:

- Pin length: 32 inches
- Maximum linear heating rate: 14.4 KW/ft
- Coolant inlet temperature: 550°F
- Coolant temperature rise across core: 518°F

Figure 11 shows the coolant outlet temperature for instantaneous flow reductions between 50 and 90%. In contrast with the hottest fuel pin of Table I, the flow reduction to produce a coolant temperature of 1620°F is 52% instead of 63%. The results show that a simultaneous 80% flow reduction and scram give a maximum coolant temperature of over 1700°F, whereas a 70% flow reduction with a 100-msec time delay before scram results in a maximum temperature below 1600°F. Therefore, it appears that the time delay involved in detecting the loss of flow and in the scram circuit before release of the rods is important only in the narrow range between 70 and 80% instantaneous flow reduction.

Digital simulation studies of the reactor and heat transport system have been carried out for a 2-ft² pipe break in one of the three primary loops. The break is assumed to occur downstream of the pump, but upstream of the check valve, and is relatively large with an opening larger than the cross-sectional area of the 18-in. pipe which ruptures. The following sequence of events is assumed to take place:

- The break occurs at 2 sec and is instantaneous to the full 2-ft² size
- The reactor is scrammed at 4 sec with a rod acceleration of 1 g and a safety system worth of 15$
- Flow coastdowns on all three pumps are started at 5 sec
FIGURE 11. Coolant Temperature Response to Instantaneous Flow Reduction: 14.4 kW/ft Hottest Fuel Pin
The check valve in the broken loop starts to close at 2.05 sec, the instant of flow reversal. The check valve is fully closed at 5.05 sec, the 3-sec closing time resulting from a dashpot.

Figure 12 shows some of the reactor temperatures after the pipe break occurs. The results show that coolant boiling will not occur if the system operates as described above.

3. Complete Flow Blockage at Reactor Startup

A completely blocked subassembly at reactor startup is of interest because this type of blockage has occurred in a fast reactor causing a fuel meltdown. Although the loss of flow would certainly be detected by a flowmeter in the subassembly, the value of the coolant thermocouple located above the fuel bundle is less obvious. If the loss of flow is not complete, then the change would eventually be detected by the thermocouple as an increase in the coolant temperature rise across the core. For a complete loss of flow, the effect on the coolant thermocouple will depend on the relative radial heat transfer between the adjacent unblocked subassemblies and axial conduction up the blocked subassembly.

Two models were used for determining the temperatures in a completely blocked subassembly. Figure 13 shows the nodal structure in which only radial heat transfer is considered. The heat generated in the fuel bundle is conducted across a 0.100-in. gap of stagnant sodium into the adjacent unblocked subassembly where it is assumed that the temperature of the flowing sodium is 600°F. The maximum steady-state sodium temperatures in the center of the blocked subassembly (node 101) were determined for several power levels:

<table>
<thead>
<tr>
<th>Reactor Power, MW*</th>
<th>Maximum Sodium Temperatures, °F</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.12</td>
<td>854</td>
</tr>
<tr>
<td>2.24</td>
<td>1124</td>
</tr>
<tr>
<td>4.48</td>
<td>1774</td>
</tr>
</tbody>
</table>

*Including ambient decay heat production
CONDITIONS:
1. 2 sec eddy current clutch time constant.
2. 1 g, 15$ scram.
3. Main flow controller setpoints at 1385 lb/sec.
4. Emergency pump speed controller setpoints at 100 rpm.
5. De-energize eddy current clutches for coastdowns but re-energizes at 300 rpm pump speeds for emergency flow control.

Average Fuel Temp.

Total Power

Hot Channel Core Outlet

Vessel Outlet Temp.

FIGURE 12. Reactor Temperatures from Simulation Studies of Pipe Break
Nodal Connections

Coolant Temperatures in °F for Reactor Power = 4.48 MW

FIGURE 13. Model for Radial Heat Transfer in Blocked Fuel Subassembly
Thus, the radial heat transfer model predicts that excessive sodium temperatures are reached at a power level of approximately 4 MW.

A model taking into account both radial and axial heat transfer is shown in Figure 14. Although the nodal structure is less detailed than the radial model, there are eleven 6-in. axial segments. A uniform radial, and chopped-cosine axial power distribution was assumed for the core region. Two cases were considered for the stagnant sodium gap between subassemblies. The first case used a constant gap of 0.100 in. and the second used a divergent gap width assuming a clearance of 0.030 in. at the bottom of the core and a dispersion of 0.0157 inch-per-inch of axial length. Sodium with an inlet temperature of 550°F and a velocity of 26 ft/sec was assumed as the normal flow conditions in the unblocked subassembly.

The maximum steady-state sodium temperatures were determined for several power levels, the power arising in decay heat as well as the fission process.

<table>
<thead>
<tr>
<th>Reactor Power, MW</th>
<th>Maximum Sodium Temperature, °F</th>
<th>Constant Gap</th>
<th>Divergent Gap</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>1155</td>
<td>1172</td>
<td></td>
</tr>
<tr>
<td>2</td>
<td>1759</td>
<td>1795</td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>2364</td>
<td>2418</td>
<td></td>
</tr>
</tbody>
</table>

The maximum sodium temperatures are actually for a square subassembly and, therefore, are much higher than the results of the radial model.

Also shown in Figure 14 is the sodium temperature distribution using the divergent gap between subassemblies for a reactor power of 2 MW. The results show that the mechanism of heat transfer is predominantly radial, toward the adjacent unblocked subassembly, and that a thermocouple located above the fuel bundle would not be able to detect the complete flow blockage.

4. Local Blockage Within Fuel Bundle - Local Coolant Temperatures

Figure 15 shows the model used for analyzing local blockages within the fuel bundle of one and six adjacent subchannels. The single subchannel
Divergent Gap Between Subassemblies

Coolant Temperatures in °F for Reactor Power = 2 MW

<p>| | | |</p>
<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>565</td>
<td>565</td>
<td>565</td>
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<td>565</td>
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<td>565</td>
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<td>567</td>
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<tr>
<td>577</td>
<td>568</td>
<td>565</td>
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<tr>
<td>659</td>
<td>585</td>
<td>565</td>
</tr>
<tr>
<td>1319</td>
<td>730</td>
<td>565</td>
</tr>
<tr>
<td>1657</td>
<td>794</td>
<td>563</td>
</tr>
<tr>
<td>1795</td>
<td>810</td>
<td>560</td>
</tr>
<tr>
<td>1777</td>
<td>791</td>
<td>557</td>
</tr>
<tr>
<td>1604</td>
<td>745</td>
<td>554</td>
</tr>
<tr>
<td>1339</td>
<td>685</td>
<td>552</td>
</tr>
</tbody>
</table>

30 in. Plenum
36 in. Core

FIGURE 14. Model for Radial and Axial Heat Transfer in Blocked Fuel Subassembly
FIGURE 16. Model for Analysis of Coolant Subchannel Blockages
blockage was simulated by zero inlet flow rate in the coolant subchannel at the $60^\circ$ apex of the triangle, and the six-subchannel blockage corresponds to zero inlet flow at the $30^\circ$ apex. For all of the cases studied, the triangle was found to be sufficiently large so that a disturbance at one apex of the triangle had a negligible effect on the coolant temperature and flow at the other apexes.

The 0.250-in. OD fuel pin were assumed to consist of 3 ft of active core and 3 ft of fission gas plenum with the following parameters:

- Pin average linear power: 13.6 KW/ft
- Radial power peaking factor: 1.0
- Axial power peaking factor: 1.24
- Sodium mass velocity: $7.62 \times 10^6 \text{lb/hr-ft}^2$

Two types of blockages were considered: blockages along the entire length of the fuel bundle and blockages which occur at some axial location with the fuel bundle.

The analyses were carried out using COBRA\textsuperscript{(7)}, a digital computer program which analyzes heat transfer and turbulent mixing in the coolant for rod bundle geometry. The thermal mixing between adjacent coolant subchannels is divided into two types. The first type is due to the net convective transport caused by flow redistribution resulting from equilizing pressures. This type is calculated from the momentum and energy equations. The second mechanism is the fluctuating (no net imbalance of flow when averaged over a short period of time) cross flow due to turbulent interchange.

This fluctuating cross flow is characterized by a parameter $\beta$ called the turbulent mixing parameter. Experiments with water using an electrically-heated test section\textsuperscript{(8)} have yielded the following correlation for the turbulent mixing parameter:

$$\beta = 0.0062 \frac{D}{S} (N_R e)^{-0.1}$$
where \( D \) is the pin OD, \( S \) is the pin spacing, and \( N_{Re} \) is the Reynolds number. For the flow blockage model and parameters, \( \beta \) is approximately 0.01, and this value is used to obtain a conservative estimate of the amount of turbulent mixing. To account for the increased mixing caused by the spiral wire wrap spacers, the turbulent mixing parameter was arbitrarily increased by a factor of four to 0.04.

**a. Blockage Along Entire Length of Fuel Bundle**

Although blockages along the entire length of the fuel bundle are unrealistic, they are an extreme case which can be compared with previous analyses. For the single-subchannel blockage, the heat which would normally flow into the blocked subchannel must flow radially and circumferentially through the fuel pin into the adjacent coolant subchannel. Figure 16 shows the axial temperature profile for the coolant in this adjacent subchannel for no turbulent mixing (\( \beta = 0 \)) and for conservative (\( \beta = 0.01 \)) and optimistic (\( \beta = 0.04 \)) values of the turbulent mixing parameter. At the end of the fission gas plenum, the coolant temperature in the subchannel adjacent to the blockage is 149°F higher for no turbulent mixing, 26°F above normal for conservative mixing, and 8°F above normal for optimistic mixing.

For the six-subchannel blockage along the entire length of the fuel bundle, the coolant temperature in the adjacent subchannels are substantially higher as shown in Figure 17. The temperatures above normal are 102°F and 36°F for turbulent mixing parameters of 0.01 and 0.04 respectively.

**b. Blockage at Specific Axial Location**

Local blockages would most probably occur at the bottom of the fuel bundle, where there is a sudden reduction in coolant flow area. However, the most severe case is a local blockage at mid-core because of the much higher power generation and, therefore, most of the analyses were carried out at mid-core. The blockage was simulated by restarting the no blockage case with zero inlet flow rate in the blocked subchannel. Thus, the blockage can be assumed to be infinitely thin.
FIGURE 16. Effect of Turbulent Mixing on Coolant Temperatures for Single Subchannel Blocked Along Entire Length of Pin Bundle
**FIGURE 17.** Effect of Turbulent Mixing on Coolant Temperatures for Six Subchannels Blocked Along Entire Length of Pin Bundle
Figure 18 shows that the single-subchannel blockage at mid-core is virtually undetectable. At the end of the fission gas plenum, the coolant temperature in the blocked subchannel is $5^\circ F$ and $1^\circ F$ above normal for turbulent mixing parameters at 0.01 and 0.04 respectively.

The blockage of six subchannels at mid-core produces a sharp coolant temperature peak in the immediate vicinity of the blockage. In Figure 19, it is seen that this peaking is substantially reduced from $368^\circ F$ to $96^\circ F$ by increasing the turbulent mixing parameter from 0.01 to 0.04. These temperature peaks would probably be much less severe if conduction within the coolant could have been taken into account.

The temperature peak is virtually eliminated by allowing a small amount of flow through the blockage, as shown in Figure 20. For the 99% blockage, which corresponds to allowing 1% of the normal flow in the blocked subchannel, the coolant temperature peak is reduced from $368^\circ F$ to $47^\circ F$ above the normal level.

The coolant temperature in the six-blocked subchannels at the end of the fission gas plenum are somewhat higher than for the single-subchannel blockage, as shown in Figure 21. The temperatures are $32^\circ F$ and $8^\circ F$ for turbulent mixing parameters of 0.01 and 0.04 respectively.

Since the flow blockage studies were carried out for a limited size geometry and assuming a uniform power profile, it was decided to analyze the six-subchannel blockage using a 1/12-segment of the fuel subassembly, as shown in Figure 22. The fuel bundle consists of 32 in. of active fuel, 6 in. of upper axial reflector, and 24 in. of fission gas plenum with an average pin linear power of 9.7 KW/ft. The coolant temperature rise across the core was $398^\circ F$. The subassembly duct shape changes from hexagonal to circular at 16 in. from the end of the fission gas plenum.

Figure 23 shows the results for complete flow blockage of six adjacent subchannels in the center of the fuel bundle, both at the inlet and at mid-core for a turbulent mixing parameter of 0.01. It is seen that the coolant
FIGURE 18. Effect of Turbulent Mixing on Coolant Temperatures in Single Subchannel Completely Blocked at Mid-Core
\[ \beta = 0 \text{ (No Turbulent Mixing)} \]

\[
\begin{array}{c}
\text{Temperature, °F} \\
1100 \\
1000 \\
900 \\
800 \\
700 \\
\end{array}
\]

\[
\begin{array}{c}
\text{Axial Distance Above Blockage, in.} \\
0.05 \\
0.10 \\
0.15 \\
\end{array}
\]

FIGURE 19. Effect of Turbulent Mixing on Coolant Temperatures in Six Subchannels Completely Blocked at Mid-Core. Detail in Region of the Blockage
FIGURE 20. Effect of Degree of Blockage on Coolant Temperatures in Six Subchannels Blocked at Mid-Core
FIGURE 21. Effect of Turbulent Mixing on Coolant Temperatures in Six Subchannels Blocked at Mid-Core
FIGURE 22. Models for Analysis of Coolant Flow and Temperature Distribution in 217 Pin Bundle
Blockage of Six Adjacent Subchannels in Center of Pin Bundle at Inlet

Blockage of Six Adjacent Subchannels in Center of Pin Bundle at Midcore

No Blockage

30° Degree Segment Model
Turbulent Mixing Parameter = 0.01

No Change in Subassembly Duct Shape

Transition from Hexagonal to Circular Duct Shape at 14 in. Above Core

FIGURE 23. Effect of Local Blockage on Temperature Distribution at End of 317 Pin Bundle
temperature in the center of the fuel bundle at the end of the fission gas plenum is approximately the same regardless of whether the blockage is at the inlet or at mid-core, and regardless of whether there is a change in the subassembly duct shape. In all the cases, a temperature rise of approximately 31°F would be expected.

5. Local Blockage Within Fuel Bundle — Local Fuel and Clad Temperatures

Figure 24 shows the model used for determining the temperature distribution inside the fuel pin for a single-subchannel blockage. A heating rate of 15.1 KW/ft was used, corresponding to the hottest fuel pin at mid-core. It was assumed that the thermal conductivity of the blockage material was zero since this condition produces the maximum fuel and clad temperatures for a non-heat generating blockage material.

The maximum fuel (node 1) and clad (node 102) temperatures as a function of time are shown in Figure 25 for a coolant temperature of 732°F. The maximum fuel temperature rises by only 156°F but the maximum clad temperature increases by over 900°F above the normal level. Although the clad does not reach its melting point, the lower strength at the high temperatures would eventually result in rupture of the clad.

The six-subchannel blockage, assuming a zero thermal conductivity for the blockage material, results in the heat generated in the blockage region being transferred axially. Figure 26 shows the nodal structure for a 0.100-in.-long blockage. The coolant temperature distribution used above the blockage was taken from Figure 19 for a turbulent mixing parameter of 0.01. Figure 27 shows the steady-state axial temperature profile in the clad for blockages of 0.050, 0.100, and 0.200 inches. The corresponding maximum clad temperatures are 1210°F, 1560°F, and 2400°F, respectively. The transient behavior of the fuel and clad temperatures is similar to that for the single-subchannel blockage in Figure 24. For example, 88% of the maximum clad temperature increase is attained in the first second for the 0.100-in. blockage.
FIGURE 24. Model for Analyzing Fuel and Cladding Temperatures for Single Coolant Subchannel Blockage
FIGURE 25. Maximum Fuel and Cladding Temperatures for Single Coolant Subchannel Blockage
FIGURE 26. Model for Analysis of Fuel and Clad Temperatures in Six-Subchannel Blockage
FIGURE 27. Steady-State Axial Clad Temperature Profile for Six-Subchannel Blockage
6. **Effect of Local Blockage on Overall Subassembly Flow**

Local blockages within the fuel bundle will increase the pressure drop through the subassembly for the same flow rate or, for a constant pressure drop across the subassembly, decrease the overall mass flow resulting in an increase in the coolant temperature rise. The decrease in flow as a function of the number of coolant subchannels blocked was determined by conservatively assuming that the blockage occurred over the entire length of the fuel bundle. A blockage which occurs at a specific axial location would be expected to result in less flow resistance.

A detailed pressure drop analysis for the driver fuel subassembly (9) was used as the basis for the analysis. The model, shown in Figure 28, divides the pressure drop into seven regions. The effect of the blocked subchannels on the coolant flow area and wetted perimeter in the pin region was used to calculate the increased resistance to flow. Keeping the pressure drop across the subassembly constant, the reduced flow rate was calculated as a function of the number of blocked subchannels. The results are presented in Table III for blockages of 1, 2, 4, 6, and 24 subchannels.

<table>
<thead>
<tr>
<th>Number of Blocked Subchannels</th>
<th>Decrease in Flow (%)</th>
<th>Coolant ΔT °F</th>
<th>Increase in Coolant ΔT °F</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>0</td>
<td>350</td>
<td>0</td>
</tr>
<tr>
<td>1</td>
<td>0.13</td>
<td>350.5</td>
<td>0.5</td>
</tr>
<tr>
<td>2</td>
<td>0.26</td>
<td>350.9</td>
<td>0.9</td>
</tr>
<tr>
<td>4</td>
<td>0.52</td>
<td>351.8</td>
<td>1.8</td>
</tr>
<tr>
<td>6</td>
<td>0.78</td>
<td>352.8</td>
<td>2.8</td>
</tr>
<tr>
<td>24</td>
<td>3.16</td>
<td>361.4</td>
<td>11.4</td>
</tr>
</tbody>
</table>
FIGURE 28. Model for Analysis of Subassembly Pressure Drop
The extent of the various blockages is shown in Figure 29. It is doubtful whether much larger blockages could be realistically postulated since the blockage would then probably occur at the subassembly inlet.

7. **Flow Reduction Due to Fuel Swelling**

Swelling of the clad reduces the cross-sectional flow area of the tube and thus increases the core pressure drop. Calculations have shown\(^{(10)}\) that a uniform diametral increase of 1.5%, occurring at an exposure of about 60,000 MWd/T or a burnup of seven at.%, will result in a reduction in the flow rate of about 5%. For a 350°F coolant core temperature rise, this corresponds to a temperature increase of 18°F.

C. **Failed Element Detection and Location**

1. **Fault Progression Considerations**

Present understanding of fuel integrity is insufficient to permit operation of the FTR following an indication of failure of a single fuel pin. Such a failure may only involve the intermittent release of small bubbles of fission gas from a fuel pin plenum. At this point, however, we are unable to rule out the possibility of the malfunction increasing in magnitude, and must, in fact, presume this to be the ordinary case.

In typical fashion, for such a model, the cladding defect will become larger, permitting a burst release of fission gas and depressurization of the pin. For defects in the fueled section, release of some fuel debris into the coolant stream is taken to be the probable result, this because of leaching or some sodium logging. With logging, the release of fuel particulate may be accompanied by the violent expulsion of sodium vapor. Poor heat transfer due to large bubbles of such vapor and gas may result in some further failure downstream. Some of the fuel debris may become lodged in the pin bundle (40 mil spacing) and promote subchannel plugging and flow blockage. Coolant stagnation downstream of these sites and a general flow reduction will increase coolant temperatures and give rise to excessive clad temperatures.
FIGURE 29. Types of Subchannel Blockages Within the Fuel Bundle
The detection of an increased bulk coolant temperature in the duct and successful safety system response is possible for blockages of 70% and possibly as high as 80%. Without safety system protection, a rapid, uniform blockage of 50% may result in sodium boiling, neglecting superheating.

For possible failed-fuel-induced blockages which range higher than 70 or 80%, unable to be sensed quickly enough with thermocouples and perhaps not reliably detected by the single duct flowmeter, some protection is necessary if possible. A PEDAL system is required to furnish that protection by anticipating such a chain of events in its early progress and limiting the occurrence to the incipient stages. If such a potential sequence of fuel failure causality is to be presumed, the PEDAL system reduces the possibility of extensive fuel damage and contamination of the primary system.

2. Basis

The basis for providing a PEDAL system in the FTR design is the inability without such a system to completely protect the reactor integrity against substantial damage by plausible occurrences growing out of failure of a single fuel pin. It is required, therefore, that cover gas samples in the vessel and primary heat exchanger be continuously monitored to detect the initiation of such a failure, notably, a substantial depressurization of a fuel pin during reactor operation.

Means for locating failure need not rely upon monitoring sodium samples for delayed neutrons if gas disengagement methods are able to be developed to a sufficient degree of efficiency and reliability. In any event it is uncertain whether a pin-depressurizing gas burst, at steady operation, will release material which contains fresh enough halogen delayed-neutron precursors in useful quantity. Thermal cycling of the fuel following failure may furnish a strengthened opportunity for use of this method.

It is requisite that the design provide the means for continuous collection of fission gas; this assumes the capability to disengage and accumulate
a sufficiently high fraction of a burst release that clad failure of a single pin can be traced to the offending subassembly. If controlled reactor shutdown is required upon detection of cover gas activity, location by gas release of the faulty piece need be accomplished in a time interval no more critically determined than the times associated with refueling.

3. Pin Depressurization

It may be that the most faithful indication that cladding failure has occurred is available in phenomena signaling pin depressurization; an ipso facto consequence of depressurization is the release of gaseous species. Thus, while artificially planted gas tags fall in this category, solid or liquid tags or fission products do not. For gaseous tags, however, their presence in a pin-depressurizing gas leak is not a physical necessity (guaranteed) as is the presence of gaseous fission products, which accumulate without doubt whenever the phenomenon of fission occurs to any extent.

Of course, for fission products, their concentration is not able to be arbitrarily and positively assured either in terms of radioactivity or gas volume. The isotopes with short half-lives have concentrations which saturate quickly, but are limited in quantity by the on-going power level. Long-lived species eventually will come to comprise most of the pressure-contributing gas component in the pin but are formed in amounts which depend upon exposure. The problem of disengaging active gas components from the coolant stream becomes relaxed with exposure in the respect that the inert gas species build up and furnish the potential of a higher volume total gas release, serving as a more effectively disengaged carrier for the active species.

In the manufacture of the fuel pins, some sorbed gases and $\text{H}_2\text{O}$ will be tolerated; upon heatup to operating temperature these gases will contribute a significant pressure component, along with that due to backfill helium. The helium backfill will facilitate leak checking in vacuo, its
pressure fixed at about 1 atm during inert atmosphere welding. At operating temperature the pin will have become initially pressurized to some 200 psi* from these sources, the pressure increasing nearly linearly as fission gases build in.

Near goal exposure, a gas pressure of about 1000 psi would be computed if the whole inventory of gaseous fission products was assumed to be present in the pin plenum, along with the initial pressure-producing components. For a pessimistic approach to cladding design this higher value would be assumed plus that arising in some overpower factor. Computations relating to gas release detection should exert pessimism in the other direction, however, and presume that only a fraction, say one-fifth, of the fission gas is a pressurizing factor, the rest being trapped at grain boundaries, etc. This view would expect a pin pressure during operation at goal exposure nearer to 500 psi.

The potential for gas release to the coolant depends also upon the retarding effect of local coolant pressure, which increases from 35 to 125 psig from the top of pin plenum to the bottom of the fueled section. Of course, during initial operation with lower pin pressures, this consideration has more importance. If initial pin pressures are much less than 200 psi, gas release may be substantially resisted due to local coolant pressure. For purposes of released-gas disengagement, non-active carrier gas volumes are also low at this time due to the slow buildup of the long half-life inert species. At only 1500 MWd/T, however, the activities of noble gases, halogens and volatile solids will have reached about one-half their expected values at goal exposure. Thus the specific activity of any gas released is highest during the early stages of pin exposure but the likelihood of sizeable release, and therefore successful disengagement and detection, is lowest then for a given cladding fault.

*This presumes the initial condition of 1 atm of helium and 0.05 cc of sorbed gas/g of fuel, both at STP, and 30 ppm H₂O
4. Noble Gas in FTR Fuel Pins

The noble gases (krypton and xenon) are expected to be the only fission products that can escape to the reactor cover gas in the event of failure of fuel cladding. Detection of these gases in the cover gas provides one potential method of determining the existence and location of a fuel cladding failure. The equilibrium production rate and inventory of each isotope of krypton and xenon in an FTR pin is summarized in Table IV together with half-life and beta and gamma decay energy.

The tabulated values assume that the FTR will operate at 400 MW, and that each operating cycle will consist of 80 days of operation at power followed by a 26-day shutdown. Goal exposure will be reached after four operating periods. It is further assumed that there will be 80 effective fueled assemblies in the core, each of which contains 217 individual pins. Thus, an average fuel pin generates 0.023 MW. In a plutonium-fueled fast reactor system, it has been assumed that 215 MeV will result per fission event.

The fission product inventories were generated using the computer code RIBD.\(^{(11)}\) Fission-product-yield factors for plutonium are those developed by C. A. Anderson and reported in LA-3383.\(^{(12)}\) Neutron capture cross-sections of the fission products are FTR spectrum-weighted values generated by R. E. Schenter and reported in BNWL-638.\(^{(13)}\) The flux level was assumed to be $5 \times 10^{15} \text{n/cm}^2\text{sec}$.

5. Delayed Neutrons from Fuel Cladding Failure

During steady operation and prior to a possible cladding failure, an equilibrium inventory of delayed-neutron-emitting isotopes will exist within each pin. The potential exists for some of this material to be released in a burst over a short period of time should failure occur. If the reactor continues to operate thereafter at a steady power level, these isotopes will continue to be generated and may be released at a constant rate. It may be assumed that the latter steady release is composed of freshly-produced
### TABLE IV

**Noble Gas in PFR Fuel Pin**

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Equil. Prod. Rate, atoms/sec</th>
<th>Exposure with No Decay</th>
<th>Inventory at Goal</th>
<th>Decay Constant</th>
<th>E&lt;sub&gt;β&lt;/sub&gt;, MeV</th>
<th>Avg. E&lt;sub&gt;β&lt;/sub&gt;, MeV</th>
<th>Total E&lt;sub&gt;γ&lt;/sub&gt;, MeV</th>
</tr>
</thead>
<tbody>
<tr>
<td>Kr 80</td>
<td>0.00</td>
<td>1.296 - 08</td>
<td>0.00</td>
<td>0</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>81</td>
<td>0.00</td>
<td>4.469 - 09</td>
<td>0.00</td>
<td>0</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>82</td>
<td>1.739 + 12</td>
<td>1.040 - 02</td>
<td>0.00</td>
<td>0</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
</tr>
<tr>
<td>83*</td>
<td>2.542 + 12</td>
<td>7.012 + 01</td>
<td>3.421 - 04</td>
<td>1.86 hr</td>
<td>1.033 - 04</td>
<td>0.00</td>
<td>0.0322 100 0.0415</td>
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<tr>
<td>83</td>
<td>0.00</td>
<td>2.563 - 00</td>
<td>0.00</td>
<td>0</td>
<td>0.00</td>
<td>0.00</td>
<td>0.00</td>
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<tr>
<td>84</td>
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<td>3.746 - 00</td>
<td>0.00</td>
<td>0</td>
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<td>85*</td>
<td>4.596 + 12</td>
<td>1.256 + 02</td>
<td>3.949 - 03</td>
<td>4.39 hr</td>
<td>4.39 - 05</td>
<td>0.835 80</td>
<td>0.233 0.305 0.20 0.414</td>
</tr>
<tr>
<td>85</td>
<td>0.00</td>
<td>8.958 - 01</td>
<td>5.93 x 10&lt;sup&gt;3&lt;/sup&gt; d 2.04 - 09</td>
<td>0.695 99+ 0.232</td>
<td>0.00</td>
<td>0.00</td>
<td></td>
</tr>
<tr>
<td>86</td>
<td>0.021 + 12</td>
<td>6.243 - 00</td>
<td>0.00</td>
<td>0</td>
<td>0.00</td>
<td>0.00</td>
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<td>87</td>
<td>7.66 + 12</td>
<td>1.895 - 03</td>
<td>1.27 hr 1.516 - 04</td>
<td>1.27 25</td>
<td>1.01 2.3</td>
<td>&lt;25 1.57</td>
<td></td>
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<tr>
<td>Kr 88</td>
<td>9.312 + 12</td>
<td>5.101 - 03</td>
<td>2.61 hr 6.85 - 05</td>
<td>0.52 70</td>
<td>0.331 2.40</td>
<td>35 2.07</td>
<td></td>
</tr>
<tr>
<td>89</td>
<td>1.069 + 13</td>
<td>2.929 + 02</td>
<td>1.116 - 04</td>
<td>3.2 min 3.61 - 03</td>
<td>4.0 100</td>
<td>1.3</td>
<td>0 0 0</td>
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<tr>
<td>90</td>
<td>9.78 + 12</td>
<td>2.697 + 02</td>
<td>1.767 - 05</td>
<td>53. sec 2.1 - 02</td>
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<td>1.07</td>
<td>0 0 0</td>
</tr>
<tr>
<td>91</td>
<td>7.165 + 12</td>
<td>1.974 + 02</td>
<td>3.918 - 06</td>
<td>10. sec 6.93 - 02</td>
<td>3.6 100</td>
<td>1.20</td>
<td>0 0 0</td>
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<tr>
<td>92</td>
<td>3.653 + 12</td>
<td>1.019 + 02</td>
<td>6.066 - 07</td>
<td>3. sec 2.31 - 01</td>
<td>5.0 100</td>
<td>1.67</td>
<td>0 0 0</td>
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<tr>
<td>Kr 93</td>
<td>1.231 + 12</td>
<td>3.454 + 01</td>
<td>1.388 - 07</td>
<td>2.0 sec 3.46 - 01</td>
<td>8.0 100</td>
<td>2.66</td>
<td>0 0 0</td>
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<tr>
<td>94</td>
<td>2.670 + 11</td>
<td>8.050 + 00</td>
<td>2.249 - 08</td>
<td>1.4 sec 4.95 - 01</td>
<td>6.4 100</td>
<td>2.14</td>
<td>0 0 0</td>
</tr>
<tr>
<td>95</td>
<td>3.345 + 10</td>
<td>5.875 + 00</td>
<td>1.166 - 08</td>
<td>1.0 sec 6.93 - 01</td>
<td>9.3 100</td>
<td>3.1</td>
<td>0 0 0</td>
</tr>
<tr>
<td>97</td>
<td>3.756 + 00</td>
<td>7.415 - 09</td>
<td>1.6 sec 6.93 - 01</td>
<td>10.6 100</td>
<td>3.5 0 0</td>
<td>0 0 0</td>
<td></td>
</tr>
<tr>
<td>Total Kr</td>
<td>1.576 + 03</td>
<td>1.347 + 01</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0.028 7</td>
</tr>
<tr>
<td>Isotope</td>
<td>Equil. Prod. Rate, atoms/sec</td>
<td>Exposure with No Decay</td>
<td>Inventory at Goal</td>
<td>Decay</td>
<td>$\beta$ Particle</td>
<td>$\gamma$ Ray</td>
<td>Total $\gamma$, MeV</td>
</tr>
<tr>
<td>---------</td>
<td>-----------------------------</td>
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<td>-------</td>
<td>-----------------</td>
<td>-------------</td>
<td>-----------------</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td>Curies</td>
<td>Volume, cm$^3$</td>
<td>Half-Life</td>
<td>$\lambda$(sec$^{-1}$)</td>
<td>$E_b$, MeV</td>
</tr>
<tr>
<td>Xe 128</td>
<td></td>
<td></td>
<td>0.00</td>
<td>8.943 - 02</td>
<td></td>
<td></td>
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<td>129</td>
<td>8.028 + 12</td>
<td></td>
<td>0.00</td>
<td>9.123 - 04</td>
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<td></td>
<td>0</td>
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<tr>
<td>130</td>
<td>1.572 + 13</td>
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<td>0.00</td>
<td>2.260 - 01</td>
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<td>0</td>
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<tr>
<td>131*</td>
<td>1.365 + 11</td>
<td>4.194 + 00</td>
<td>8.631 - 03</td>
<td>12.0 d</td>
<td>6.69 - 07</td>
<td>0</td>
<td>0</td>
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<td>2.199 + 01</td>
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<td>3.044 + 13</td>
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<td>0.00</td>
<td>3.109 + 01</td>
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<td>0</td>
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<td>133*</td>
<td>1.003 + 12</td>
<td>3.019 + 01</td>
<td>1.191 - 02</td>
<td>2.3 d</td>
<td>3.49 - 06</td>
<td>0</td>
<td>0</td>
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<tr>
<td>133</td>
<td>3.813 + 13</td>
<td>1.037 + 03</td>
<td>9.371 - 01</td>
<td>5.27 d</td>
<td>1.52 - 06</td>
<td>0.345</td>
<td>100</td>
</tr>
<tr>
<td>134</td>
<td>4.314 + 13</td>
<td></td>
<td>0.00</td>
<td>4.422 + 01</td>
<td></td>
<td></td>
<td>0</td>
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<tr>
<td>Xe 135*</td>
<td>1.446 + 13</td>
<td>3.628 + 02</td>
<td>6.873 - 04</td>
<td>16 min</td>
<td>7.23 - 04</td>
<td>0</td>
<td>0</td>
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<tr>
<td>135</td>
<td>3.132 + 13</td>
<td>1.242 + 03</td>
<td>8.163 - 02</td>
<td>9.19 hr</td>
<td>2.1 - 05</td>
<td>0.548</td>
<td>5</td>
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<tr>
<td>136</td>
<td></td>
<td></td>
<td>0.00</td>
<td>4.722 + 01</td>
<td></td>
<td></td>
<td>0</td>
</tr>
<tr>
<td>137</td>
<td></td>
<td></td>
<td>1.174 + 03</td>
<td>5.453 - 04</td>
<td></td>
<td></td>
<td>0</td>
</tr>
<tr>
<td>138</td>
<td>3.323 + 13</td>
<td>8.974 + 02</td>
<td>1.497 - 03</td>
<td>14 min</td>
<td>8.25 - 04</td>
<td>3.02</td>
<td>100</td>
</tr>
<tr>
<td>139</td>
<td>1.998 + 13</td>
<td>5.417 + 02</td>
<td>4.449 - 05</td>
<td>41 sec</td>
<td>1.69 - 02</td>
<td>5.15</td>
<td>100</td>
</tr>
<tr>
<td>140</td>
<td>9.653 + 12</td>
<td>2.639 + 02</td>
<td>8.358 - 06</td>
<td>16 sec</td>
<td>4.33 - 02</td>
<td>3.97</td>
<td>100</td>
</tr>
<tr>
<td>141</td>
<td>3.412 + 12</td>
<td>9.405 + 01</td>
<td>3.174 - 07</td>
<td>1.7 sec</td>
<td>4.07 - 01</td>
<td>6.07</td>
<td>100</td>
</tr>
<tr>
<td>142</td>
<td>7.426 + 11</td>
<td>2.040 + 01</td>
<td>6.075 - 08</td>
<td>1.5 sec</td>
<td>4.62 - 01</td>
<td></td>
<td></td>
</tr>
<tr>
<td>143</td>
<td>8.697 + 10</td>
<td>7.151 + 00</td>
<td>1.419 - 08</td>
<td>1.0 sec</td>
<td>6.93 - 01</td>
<td>6.95</td>
<td>100</td>
</tr>
<tr>
<td>144</td>
<td>4.514 + 00</td>
<td>8.958 - 09</td>
<td>1.0 sec</td>
<td>6.93 - 01</td>
<td>5.78</td>
<td>100</td>
<td>1.9</td>
</tr>
<tr>
<td>Total Xe</td>
<td></td>
<td></td>
<td>5.679 + 03</td>
<td>1.459 + 02</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
species. Given the release conditions and dilution factors, it is possible to estimate the signal strength that might be achieved for specific conceptual arrangements of a delayed-neutron detector system, either for the initial burst or for a release corresponding to a continuing production.

Generation rate and equilibrium inventory in an FTR fuel pin was computed for the fission product isotopes $^{87-90}$Br and $^{137-139}$I, which are generally thought to be responsible for the delayed neutrons with half lives greater than about 1.5 seconds. Yield factors for these isotopes were based on theoretical values generated by Anderson (14) for fast neutron-induced fission of $^{239}$Pu. Values for the probability that a neutron would be emitted by a particular isotope are based on a study by Hermann, et al. (15). The latter values were chosen to provide reasonable agreement with the group-averaged valued tabulated by Keepin, et al. (16).

If all of the isotopes listed above were released from an entire pin when the cladding failed and thereafter remained physically together while being transported by the sodium coolant, the activity would decay with time according to the relation:

$$ f \sum_{i} Y_i P_i \exp(-\lambda_i t) \text{ neutrons per second}, $$

where $f$ is the fission rate per second within the whole pin, $Y_i$ is the yield of the $i$-th isotope, $P_i$ is its neutron decay probability, and $\lambda_i$ is its decay constant. The summation is taken over the seven isotopes of interest. As this material passes through the detector, the number of neutrons is given by:

$$ f \sum_{i} \frac{Y_i P_i}{\lambda_i} \left[ \exp(-\lambda_i a) - \exp(-\lambda_i b) \right] $$

where $a$ is the time at which the material enters the detector and $b$ is the time at which it leaves the detector, both times being measured from the time the material was released from the pin within the reactor. The function plotted in Figure 30 is:
FIGURE 30. Activity of Delayed Neutron Precursors (Functions for Computation)
\[
F(t) = \sum_{i} \frac{Y_i P_i}{\lambda_i} \exp(-\lambda_i t),
\]

which can be used together with release fraction and dilution factors, to determine the total number of neutrons released within the detector, or, by dividing by the time interval, the average neutron release rate during the time the material passes through the detector.

Each fission event will, on the average, produce isotopes emitting neutrons that will decay according to the relation:

\[
\sum_{i} Y_i P_i \lambda_i \exp(-\lambda_i t).
\]

For a constant fission rate, \( f \), the neutron emission rate of the stream of this material passing through the detector is

\[
f \sum_{i} Y_i P_i [\exp(-\lambda_i a) - \exp(-\lambda_i b)].
\]

The plotted function (Figure 30)

\[
F'(t) = \sum_{i} Y_i P_i \exp(-\lambda_i t)
\]

can be used as above, to determine the neutron source strength within specific detector arrangements.

6. Fission Gas Release Mechanisms Based on Experiment

Some useful estimates of fission gas release from an FTR fuel pin due to a possible cladding failure may be abstracted from the results of recent experiments dealing with the problem of fission gas evolution, migration and release.

At Pacific Northwest Laboratory, selected full length fuel pieces composing part of a high power density fuel assembly in the Plutonium
Recycle Test Reactor (PRTR) are the subject of tests of fission gas evolution (Figure 31). Pressure and temperature of the gas in a fuel pin plenum are monitored continuously. Operating conditions provide steady linear heat rates from 9 to as high as 20 kW/ft; some irradiations reached 10,000 MWd/T and are expected to go to a goal of 45,000 MWd/T. Maximum fuel temperatures are nominally 2500 to 2600°C in these Zr-2-clad, 58-in. long columns of pneumatically impacted UO₂-2 wt% PuO₂ which have been vibrationally compacted to 86% of theoretical density.

Some results obtained thus far in that work (17) are given in Figures 32 and 33. The authors conclude that, for the burnup ranges tested, fission gas release rates for packed particle, mixed-oxide fuel are predictable and directly proportional to fuel temperature. A pressure-contributing release from internal trapping sites such as grain boundaries requires a thermal cycling of the fuel, however, and no release is noted at steady operation. Typical releases are larger during shutdown (cooldown) than during a fuel temperature increase, prescinding from thermodynamic gas law effects.

Work at Oak Ridge (18) has given extensive consideration to models for gas emission and trapping using small samples of fine-grained and single-crystal UO₂ fuels. Release of xenon and krypton was found to increase exponentially for temperatures above 600°C. At lower temperatures there is little or no temperature dependence for gas release, the predominate mechanisms being recoil and knock-out of fission products, which is proportional to fission rate. Trapping of gases in internal defects may possibly occur due to recoil fragment-induced traps (19) (generated in proportion to the fission rate) as well as by holdup in grain boundary spaces, thus nullifying the effect of inventory increases.

Oscillation of the neutron flux and temperature in the sample was correlated with gas emission in order to investigate the dynamics of release in the Oak Ridge work. A release model was constructed which suggests that only a thin surface region will be involved in the fission gas release process at temperatures below 1500°C.
FIGURE 37. PRTR In-Reactor Fuel Rod Pressure Monitoring Element
Calculated Pressure Buildup using Fuel Temperature Dependent Fission Gas Release Data, and Assuming 100% Release of Sorbed Gas (0.06 cc/g) and H$_2$O (20 ppm).

Vertical Lines Relate to Reactor Power Cycles

Measured Pressure Buildup in High Burnup Fuel Rods

**FIGURE 32. Internal Pressure Buildup in High Power FRTR Fuel Rods**
FIGURE 32. Fission Gas Release Rate for FRTR Fuel Rods
Experiments at Chalk River, similar to those at Pacific Northwest Laboratory, have employed somewhat smaller fuel pieces (243 mm long) composed of stacks of sintered $\text{U}_2\text{O}_3$ pellets enriched to 2.4% by weight $^{235}\text{U}$ in total $\text{U}$. Densities of 10.25, 10.5 and 10.75 g/cm$^3$ were obtained. These were assembled into (0.45 mm wall thickness) type 304 stainless steel tubes. Published results of that work include, for various test fuel pieces, data on element burnup and power output, fission product gas release, and gas pressure buildup in time as related to operating power conditions.

Although the fuels were smaller and the exposures lower than in the Pacific Northwest Laboratory work, there is a substantial agreement where experiment specifications overlap. A noteworthy conclusion also found in these data is that very little pressure change occurs during steady operation. Also, since these fuels have linear heat rates nearly the same as expected for FTR fuels (~15 kW/ft), the values of pressure-contributing gas fractions are of interest. A useful range for specification of this fraction during operation is 0.25 to 0.40. It is difficult to associate the physical description of clad defects with gas release characteristics. However, it is useful to consider the limiting case of instantaneous pin depressurization or burst release.

Radiometallurgical postirradiation examination of cold, high power density PRTR fuels includes a cladding puncturing procedure with a 1/8-in drill to collect fission gas. For a hole this size, whether in the gas plenum region or in the fueled section of a pin, the operators have experienced a depressurization characterized as a burst rather than as a slow outgassing. Such a burst release of fission gas is expected to liberate about one-third the total inventory, which is a reasonable average of the expected range. Because it is desirable to maintain a higher degree of conservatism in the design than the burst release affords, a fraction of 10% of the liberated inventory is arbitrarily assumed.
Referring to Table IV, the activity of the total inventory of contained, noble gases in an FTR fuel pin at goal exposure is expected to be about 7500 curies and even soon after the beginning of a fuel cycle (~1500 MWD/T) nearly half that much. Some 80% of this material may be trapped at interstitial sites in the pin, depending upon operating conditions. If one-tenth of the remainder were released upon clad failure an activity of 150 curies would be released to the coolant. Therefore it is concluded from experimental information and from inventories presented in Table IV that the FEDAL system should be capable of detecting a pin failure in which a release of the order of 150 curies of noble gases ensues.

7. Delayed Neutron Precursor Release

The concentration of active delayed-neutron precursors in the fuel pin plenum is expected to be very low, just as is the case for all short half-life fission product species.* In the first place, a significant aging is expected to occur before a fission product atom finds its way to a grain boundary by diffusion or arrives there due to knock-out by a recoil fission fragment. Only direct recoil particles themselves arrive there fresh.

At steady reactor operation, grain boundary motion is not sufficiently large to produce the strains by which gases normally are released from grain boundary entrapment, as bubbles, to the larger fractures and cracks which effectively constitute an extension of the pin plenum. These latter volumes, while small, can transport a flow of gas such that it will readily vent to the coolant if the cladding fails.

There is a tendency for gases to migrate along the grain boundary trapping spaces to positions of higher fuel temperature by the vapor transfer mechanism, even while at steady reactor power. Also, the growth of columnar grains is expected to result in a sweeping action, by which bubbles are coalesced and may reach positions near some openwork gas space.

*Table V identifies delayed-neutron precursor decay components. Specific decay and abundance data for various fissile species is given in Table VI.
<table>
<thead>
<tr>
<th>Group</th>
<th>Precursor*</th>
<th>Half-Life (sec)</th>
<th>Assignment</th>
</tr>
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<td>54.5</td>
<td>Br$^{87}$</td>
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</tr>
<tr>
<td>2</td>
<td>24.4</td>
<td>$^{137}$ Br$^{88}$</td>
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</tr>
<tr>
<td>3</td>
<td>6.3</td>
<td>$^{138}$ Br$^{89}$</td>
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</tr>
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<td></td>
<td>4.4</td>
<td>$^{139}$ Br$^{90-92}$</td>
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</tr>
<tr>
<td>4</td>
<td>2.0</td>
<td>$^{139}$ Cs, Sb or Te</td>
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</tr>
<tr>
<td></td>
<td>1.6</td>
<td>$^{139}$ Br$^{90-92}$</td>
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</tr>
<tr>
<td></td>
<td>&lt;1.5</td>
<td>$^{139}$ Kr$^{93}$</td>
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<tr>
<td>5</td>
<td>0.5</td>
<td>$^{140}$ I$^{140}$ + Kr?</td>
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<tr>
<td>6</td>
<td>0.2</td>
<td>(Br, Rb, As + ?)</td>
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*Tentative or uncertain items in parenthesis*
# Table VI

Fast-Fission Delayed-Neutron Data

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<tr>
<th>Group</th>
<th>$\frac{T_1}{T_0}$ (sec)</th>
<th>Relative Abundance $\frac{a_i}{a}$</th>
<th>Absolute Group Yield (per cent) for Pure Isotope</th>
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<td>235U</td>
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<td>0.038 ± 0.003</td>
<td>0.063 ± 0.005</td>
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<tr>
<td>238U</td>
<td>0.0412 ± 0.0017</td>
<td>0.013 ± 0.001</td>
<td>0.054 ± 0.005</td>
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<tr>
<td>233U</td>
<td>0.0070 ± 0.0004</td>
<td>0.086 ± 0.003</td>
<td>0.060 ± 0.003</td>
</tr>
<tr>
<td>239Pu</td>
<td>0.0063 ± 0.0003</td>
<td>0.038 ± 0.003</td>
<td>0.024 ± 0.002</td>
</tr>
<tr>
<td>240Pu</td>
<td>0.0088 ± 0.0006</td>
<td>0.028 ± 0.003</td>
<td>0.022 ± 0.003</td>
</tr>
<tr>
<td>232Th</td>
<td>0.0496 ± 0.0020</td>
<td>0.034 ± 0.002</td>
<td>0.169 ± 0.012</td>
</tr>
</tbody>
</table>
But such mobility at steady operation is seen to occur over a period of days rather than in just a minute or two.

Thus, the gaseous neutron precursors are considerably depleted in activity by the time they reach an open gas space which communicates with the pin plenum. If a gas burst occurs, liberating typically one-third the fission product inventory, one cannot expect to find one-third the equilibrium activity of delayed-neutron precursors in the release since that quickly decaying material is more likely to be tied up at internal trapping sites. Particles which recoil directly or some freshly-produced precursors which are knocked out into open gas spaces are probably the largest contributors to neutron activity available upon clad failure.

Presumably, the more porous and fractured the fuel structure becomes, the more probably fresh halogen precursors are able to be vented to the coolant if a failure occurs. While oxide fuels are many orders of magnitude more "open" and gas-conductive than metal fuels in this respect, the quantity of fuel which is within recoil range of the surface of large fractures is still a small fraction of the total fuel material. The activity expected to characterize a typical burst release is computed below, taking into account the equilibrium activity of delayed-neutron precursors in a pin and the geometric likelihood of their being swept out to the coolant for a pin failure.

**Activity of a Delayed-Neutron Precursor Burst Release**

**Contained, unreleased activity of all fuel pin precursors, new and old.**

\[ = 3.4 \times 10^{12} \text{ @ equil. production} \]

**Fraction of fuel pin areas within average recoil range of surface/fuel pin area total.**

\[ = \frac{2\pi \delta}{\pi r^2} = \frac{2\delta}{r} \]

For \( \delta = 10\mu \)

\[ = 2 \times 10 \times 10^{-4} \]

\[ = 0.105 \times 2.54 = 1 \times 10^{-2} \]

\[ r = 0.105 \text{ inch} \]
Fraction of recoil particles in range of surface that "stop" in gas space (estimated). (knock-out of precursors and UO$_2$ particles to the gas space may compensate somewhat.)

Activity of contained, unreleased precursors, new and old, in gas space of fuel pin equilibrium.

Fraction swept out with a burst of gas.

Burst precursor activity

\[ = 3 \times 10^{-2} \]
\[ = 3.4 \times 10^{12} \times 1 \times 10^{-2} \times 3 \times 10^{-2} \]
\[ = 10^9 \]
\[ = 1/10 \]
\[ = 1 \times 10^8 \text{ neutrons/sec.} \]

It is likely that the above estimate for the source of neutrons released to the coolant is optimistic in view of the fact that migration of precursors to a local defect can be severely attenuated by closing of the gas gap in irradiated fuel. On the other hand, cracks develop in oxide fuel, providing paths for precursor migration and constituting large surface areas. To bound the source strength estimate, a lower limit involving gas depressurization may be considered in which the only exposed surface is the top surface of the uppermost pellet. This activity is of the order of $10^5$ neutrons/sec and would be released from the plenum region.

Another means of estimating the fraction of gaseous delayed-neutron precursors normally situated in the "open" gas space of a fuel pin derives from considerations of gas release rates for low fuel temperatures. It is commonly held by researchers in this field that temperature-dependent gas migration and diffusion mechanisms operate very weakly at temperatures lower than about 1000°C $\pm$ 300°C.\(^{(17,18,20)}\) At lower temperatures the release is governed by recoil (range: $\sim$10$\mu$m) and knock-out of material from a thin surface layer ($\sim$10 Å) due to the action of recoil fragments.
Production by recoil and knock-out depends upon local fission rate and is not inherently influenced by temperature increases. Therefore, for a given fission rate, the low temperature gas release rates should constitute an upper limit of the recoil plus knock-out release rates, and certainly for release due to recoil alone. Typically released gas inventory fractions for temperatures below $1000^\circ C \pm 300^\circ C$ range from $5 \times 10^{-3}$ to $1 \times 10^{-3}$.

What is to be noted from this is that the expected release fractions to the gas space given above based on geometric surface considerations may not be too low since they are only about a factor of 10 lower than these upper limit values.

For this study, the fuel surface recoil model was examined for its usefulness in estimating the magnitude and effect of delayed-neutron precursor release from a faulty fuel pin. An attempt was made to explain the count rate produced in the Fuel Element Rupture Detection (FERD) system at EBR-II due to delayed-neutron precursor release from a bare uranium metal pin inserted into the core during operation. The following steps in this analysis are given below.

a. Fission rates of the following materials are assumed for the bare pin position at reactor power = 62.5 MW:

\[
\begin{align*}
\text{U}^{235} & : 0.94 \times 10^{13} \text{ fission/g U}^{235}-\text{sec} \\
\text{U}^{238} & : 0.56 \times 10^{12} \text{ fission/g U}^{238}-\text{sec} \\
\text{Pu}^{239} \text{ (if any)} & : 1.13 \times 10^{13} \text{ fission/g Pu}^{239}-\text{sec}
\end{align*}
\]

b. Size and material composition of the bare metal pin was:

\[
\begin{align*}
\text{Total weight} & : 68 \text{ g (including 5\% fission)} \\
\text{Length} & : 14.22 \text{ in.} \\
\text{Diameter} & : 0.144 \text{ in.} \\
\text{Uranium weight} & : 64.6 \text{ g} \\
\text{U}^{235} \text{ weight} & : 33.85 \text{ g (52.4\%)} \\
\text{U}^{238} \text{ weight} & : 30.75 \text{ g}
\end{align*}
\]
c. Pin fission rate at 9.1 MW:

$$U^{235}_{\text{F.R.}} = \frac{9.1}{62.5} \times 0.94 \times 10^{13} \times 33.85 = 4.63 \times 10^{13} \frac{\text{fissions}}{\text{sec}}$$

$$U^{238}_{\text{F.R.}} = \frac{9.1}{62.5} \times 0.56 \times 10^{12} \times 30.75 = 0.251 \times 10^{13} \frac{\text{fissions}}{\text{sec}}$$

Total F.R. = $4.87 \times 10^{13}$ fission/sec.

d. The volume of the fuel pin which participates in releasing precursors to the sodium coolant is the volume of fuel in a thin skin pin surface 0.242 mils* (= 6.15μ) thick on the pin surface; only one-half of the recoil particles are released toward the coolant, however. The effective precursor-releasing fuel fraction is:

$$\frac{2\pi \cdot 0.242 \times 10^{-3} \times 0.072}{\pi \cdot (0.072)^2} = \frac{0.242 \times 10^{-3}}{0.072} = 3.36 \times 10^{-3}$$

e. The total effective precursor-releasing fission rate for the bare pin is then:

$$\text{Effective F.R.} = 4.87 \times 10^{13} \times 3.36 \times 10^{-3} = 1.636 \times 10^{11} \frac{\text{fissions}}{\text{sec}}$$

f. In the FTR, at 400 MW, 80 subassemblies, 217 pins/subassembly, there is

$$400/(80 \times 217) = 400/17,360 = 0.02304 \text{ MW/pin.}$$

at $1.60 \times 10^{-3}$ watt-sec/fission and

$$\frac{0.02304 \times 10^6 \text{ watts/pin}}{3.48 \times 10^{-11} \text{ watt-sec/fission}} = 6.62 \times 10^{14} \text{ fissions/sec per FTR fuel pin.}$$

g. The effective fission rate in the bare pin is fractionally

$$\frac{1.636 \times 10^{11}}{6.62 \times 10^{14}} = 2.47 \times 10^{-4}$$

of that in the FTR pin.

*Private communication, R. R. Smith to J. J. Regimbal, August 28, 1968
h. For the \(i^{th}\) isotope the total number of neutrons which may be expected to be released after an infinite time following its formation is

\[
Y_i P_i = A \int_{t=0}^{\infty} e^{-\lambda_i t} \, dt = \frac{A}{\lambda_i} \left[ e^{-\lambda_i t} \right]_0^{\infty} = \frac{A}{\lambda_i} - 0
\]

normalized to the precursor production associated with a single fission. This integral neutron production is therefore released in time following fission according to

\[A e^{-\lambda_i t} \text{ or } Y_i P_i \lambda_i e^{-\lambda_i t}\]

the value of \(A\) having been determined above.

If this function is summed over all \(i\) and integrated between two values of time (over an interval corresponding to the holdup time of a delayed neutron sensing chamber) since the fission event, the definite integral is the number of neutrons released in that interval per fission event. Therefore, the rate at which neutrons are released by decay in that interval is controlled by the fuel pin fission rate, normalized by the value of

\[
\int_{t=a}^{b} \Sigma_i Y_i P_i \lambda_i e^{-\lambda_i t} \, dt
\]

thus the activity in a section of coolant stream due to a continuing production of fresh precursors recoiling into that stream is, for a fission rate \(f\),

\[
\text{Activity} = f \sum_i Y_i P_i \left[ e^{-\lambda_i a} - e^{-\lambda_i b} \right]
\]

for the pipe section whose ends are \(t = a\) and \(b\) downstream from the fission and recoil site.
From a plot of
\[ f \sum Y_i \alpha_i e^{-\lambda_i t} \]
which gives the activity potential as a function of time for precursors generated at a rate of \( f \) fissions per second, one may evaluate the activity of a flowing coolant section defined by end-point times \( a \) and \( b \).

As an approximation, using the decay-of-potential-activity curves presented in Figure 30 for FTR fuel pins (note plutonium enrichment -- therefore lower \( \beta_{\text{eff}} \) by factor of 3 and different \( \lambda_{\text{eff}} \) over all groups) for a 1-ft long section of 2-in. sodium pipe (0.1-sec generation interval corresponds to 1-ft section) for a 17-sec transport time, the delayed neutron activity in the graphite stack is \( 3 \times 10^9 \) n/sec.

i. Therefore, for the bare pin the value is
\[ 3 \times 3 \times 10^9 \times 2.5 \times 10^{-4} = 2.25 \times 10^6 \text{ n/sec} \]
if all the released precursors were passing through the 2-in. pipe.

j. Only \( \frac{100 \text{ gal/min}}{9000 \text{ gal/min} \times 0.75} = 0.0148 \) of the release is in this pipe, however, or an activity of \( 2.25 \times 10^6 \times 1.48 \times 10^{-2} = 3.33 \times 10^4 \) n/sec.

k. These \( 3.33 \times 10^4 \) n/sec are fast neutrons, however, and must be thermalized, then captured in a BF_{3} detector and a count rate produced. The lumped efficiency of this process is taken to be \( 6 \times 10^{-3} \). Therefore, the count rate produced ultimately by the bare pin is
\[ 3.33 \times 10^4 \times 6 \times 10^{-3} = 200 \text{ counts/sec} \]

Approximately 60 counts/sec were observed for these conditions on EBR-II experiment. If the lumped thermalization and sensor efficiency were taken to be \( 1.8 \times 10^{-3} \), a 60 n/sec count rate would be computed. If this lumped efficiency were reduced by a 6.15\( \times 10^{-3} \) fraction to \( 1.1 \times 10^{-3} \), the
observed count rate of 60 counts/sec would be computed based on a model assuming a 10μ-thick effective precursor-producing skin on the bare pin.

Uncertainties in total pin fission rate, coolant flow and mixing, and sample transport time should be small compared to the largest potential error in the analysis, the assignment of a lumped factor for neutron thermalization and counting efficiency. This latter, assumed to have a value of $10^{-3}$, is not known well to within a factor of 10; a corresponding error may be reflected directly in the evaluation of the thickness of the fuel surface layer from which recoil release can occur.

It is concluded from analysis and experiment that a neutron source between $10^5$ and $10^8$ neutrons/sec would accompany an instantaneous pin depressurization. The largest uncertainty in attempting to narrow this range appears to be in establishing the fraction of released precursors that can migrate to a local defect and the character of that clad defect. Further experimental data bearing on these factors is needed to refine the estimates given.

8. Failure Location Method Based on Implanted Tags

This method relies on the release of a detectable quantity of labelling material prepared in advance and enclosed within the cladding along with the fuel during fabrication. The most acceptable tagging materials are the gaseous species of chemically inert isotopes. As opposed to solid or liquid forms these satisfy the criterion of guaranteed release upon pin depressurization and present a good potential for subsequent disen­
gagement from the sodium coolant.

By means of mass spectrometry, the relative amounts of a mix of isotopes of a noble gas tag may be determined and their origin traced to a unique source in a single subassembly. Relative quantities of the tag-composing isotopes would be varied from one subassembly to the next, in a fashion that all the pins in a given bundle receive the same character­istic implant.
If the primary circuit or cover gas were to become contaminated with material used for tagging, the effectiveness of the method would be reduced. For very high background levels the system could be rendered useless, thus it is imperative that possible sources of contamination be assessed in screening for candidate tagging materials. On the basis of the relative gas composition of air given below, the isotopes of xenon appear least likely to become contaminated by ingress of air, considered to be an important potential means of contamination.

COMPONENTS OF ATMOSPHERIC AIR*
(Exclusive of Water Vapor)

<table>
<thead>
<tr>
<th>Constituent</th>
<th>Content (percent)</th>
<th>Content (ppm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>N\textsubscript{2}</td>
<td>78.084 ± 0.004</td>
<td></td>
</tr>
<tr>
<td>O\textsubscript{2}</td>
<td>20.946 ± 0.002</td>
<td></td>
</tr>
<tr>
<td>CO\textsubscript{2}</td>
<td>0.033 ± 0.001</td>
<td></td>
</tr>
<tr>
<td>Ar</td>
<td>0.934 ± 0.001</td>
<td></td>
</tr>
<tr>
<td>Ne</td>
<td>18.18 ± 0.04</td>
<td></td>
</tr>
<tr>
<td>He</td>
<td>5.24 ± 0.004</td>
<td></td>
</tr>
<tr>
<td>Kr</td>
<td>1.14 ± 0.01</td>
<td></td>
</tr>
<tr>
<td>Xe</td>
<td>0.087 ± 0.001</td>
<td></td>
</tr>
<tr>
<td>H\textsubscript{2}</td>
<td>0.5</td>
<td></td>
</tr>
<tr>
<td>CH\textsubscript{4}</td>
<td>2</td>
<td></td>
</tr>
<tr>
<td>N\textsubscript{2}O</td>
<td>0.5 ± 0.1</td>
<td></td>
</tr>
</tbody>
</table>

Xenon residues in the manufacture of argon gas could also contribute contamination, as could gas from a previously failed pin.

Preliminary calculations show that the only xenon isotopes acceptable for conditions of flux and burnup in the FTR are $^{124}\text{Xe}$, $^{126}\text{Xe}$, and $^{129}\text{Xe}$. Selection of this set arises from an evaluation of physical properties that will allow expectation of an invariant, identifying code. To satisfy this criterion, the isotopes must be stable (long-lived), have a very low cumulative fission yield, and be relatively unaffected in a high neutron flux (low neutron capture cross section). These gaseous isotopes may be implanted in each fuel pin upon manufacture. Approximately 1 cc of tag material must be enclosed to provide a sufficiently strong signal to detection and analysis equipment.

Use of tags in an FTR FEDAL system are presently envisioned as a backup means for failure location. If more than a single subassembly became faulty at any one time, or if a significant amount of tramp xenon were to occupy the coolant and cover gas systems, the confused (overlapped) tag codes would point to any of a large set of element pairs which could potentially release such a mix. That is, for simultaneous failures the separate codes will be essentially obliterated.

Prediction of the gas release reliability of a tagging system suffers from the same uncertainties attending prediction of the quantity of fission gas release upon fuel failure and its rate of solubility in sodium. Moreover, the initial presence of the tag in a pin is guaranteed only by fabricating procedures whereas fission gas builds in quickly as the fuel is burned.

At the present time it is not assured that as many unique isotopic tags as there are fuel assemblies in the core can be made using present sources of $^{124}\text{Xe}$, $^{126}\text{Xe}$, and $^{129}\text{Xe}$. This is due to the unavailability of high isotopic purity. Further study of isotopic identification appears warranted.

Of the twenty-seven known Xe isotopes, only those with masses 124, 126, 128, 129, 130, 131, 132, 134 and 136 are stable. Many of the Xe isotopes are end products of the fission process. They may be produced
either directly from fission or by a chain of β-decays or by neutron capture of some species in the chain followed by a secondary chain of β-decays. Regardless of their mode of generation, it is essential that the cumulative fission yields shall not substantially alter the isotopic ratio of a tag.

To obtain a sufficient number of tags, it has been proposed that the isotopic composition of a given tag differ from any other tag by about 2%. For example, given a three-component tag, typical isotopic concentrations might be:

<table>
<thead>
<tr>
<th>Component</th>
<th>Tag No.</th>
<th>(a)</th>
<th>(b)</th>
<th>(c)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>70</td>
<td>20</td>
<td>10</td>
</tr>
<tr>
<td>(n)</td>
<td></td>
<td>70</td>
<td>18</td>
<td>12</td>
</tr>
<tr>
<td>(n + 1)</td>
<td></td>
<td>70</td>
<td>16</td>
<td>14</td>
</tr>
<tr>
<td>(n + 2)</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

Obviously, if the concentration of a single component in any tag changes by 1% of the total tag, the character of the tag will not be clearly distinguishable from an adjacent tag in the series.

The volume of Xe gas proposed for tagging is 1 cm$^3$ (STP) per fuel pin. A limit of 1% is set as the maximum acceptable cumulative fission yield for any component of a tag. The cumulative fission yield for any component of a tag, therefore, must not exceed 0.01 cm$^3$ (STP).

Table IV provides a list of the cumulative yield of Xe isotopes in cm$^3$ (STP) per fuel pin in the FTR at 4.5 x 10$^3$ MWd/T exposure. Of those isotopes qualifying in terms of stability, only $^{124}$Xe, $^{126}$Xe, and $^{129}$Xe meet the criterion of low fission yield. A cumulative fission yield of
approximately 0.09 cm$^3$ (STP) of $^{128}$Xe/fuel pin is calculated to be generated in FTR at 4.5 x $10^3$ MWD/T.

From the same argument given above, it is clear that neutron capture must not alter the isotopic ratio of a Xe tag by more than 1%.

If $N_a$ is the number of atoms of component (a) in a tag, then $\phi \sigma_a t N_a$ is the number of atoms of (a) consumed by neutron capture in flux $\phi$ in time $t$. The ratio of the burned up atoms to the total atoms of a component is

$$\frac{\phi \sigma_a t N_a}{N_a}$$

We are concerned with a change of a component amounting to 1% of the total tag. In effect, we can tolerate a large fractional burnup if the component is a minor constituent but only a small fractional burnup if the component is a major constituent. A burnup of 1% may be expressed by

$$\frac{\phi \sigma_a t N_a}{N_a} = 0.01$$

where $F_a = \text{initial fraction of the component in the tag}$. The design objective of the FTR is to operate at an average flux across the core of about 5 x $10^{15}$ n/cm$^2$-sec for an average fuel cycle of about 320 days. We can therefore state that

$$\phi = 5 \times 10^{15} \text{ n/cm}^2\text{-sec}$$
$$t = 2.77 \times 10^7 \text{ sec}$$
$$\sigma_a F_a = (10^{-2}) \frac{10^{-2}}{t} = 7.22 \times 10^{-26} \text{ cm}^2 \text{ - sec}^{-1}$$

and $0.96 \geq F_a \geq 0.02$

for a three component system. Thus the maximum tolerable cross section is
The precise value for the "maximum tolerable cross section" for a given isotope depends on $F$, its fraction of the total tag.

The above calculations ignore the change of macroscopic cross section with time as a component of the tag is burned up. The net result of this change is to narrow and shift somewhat higher the span for $\sigma$ calculated above.

Cross sections for neutron absorption in the xenon isotopes, prepared$^{(13)}$ in connection with fission product inventory calculations for FTR fuel using the RIBD code$^{(11)}$ are listed below:

<table>
<thead>
<tr>
<th>Xenon Isotope</th>
<th>Absorption Cross Section in FTR Spectrum (barns)</th>
</tr>
</thead>
<tbody>
<tr>
<td>124</td>
<td>0.13</td>
</tr>
<tr>
<td>126</td>
<td>0.06</td>
</tr>
<tr>
<td>128</td>
<td>0.230</td>
</tr>
<tr>
<td>129</td>
<td>0.32</td>
</tr>
<tr>
<td>130</td>
<td>0.086</td>
</tr>
<tr>
<td>131</td>
<td>0.291</td>
</tr>
<tr>
<td>132</td>
<td>0.067</td>
</tr>
<tr>
<td>134</td>
<td>0.047</td>
</tr>
<tr>
<td>136</td>
<td>0.012</td>
</tr>
</tbody>
</table>

From the above values for the isotopes under consideration it is concluded that the $^{126}$Xe component in a tag set may be used as a higher relative fraction than $^{124}$Xe or $^{129}$Xe, the latter being restricted to fractions less than 0.22 to limit its burnup to less than 1%.

D. Reliability Requirements for the In-Channel Instrumentation

The reliability required of the in-channel instrumentation is sensitive to safety and operating philosophy. A comparison is presented here which shows how the operating philosophy effects the in-channel instrumentation.
reliability requirements. The requirements are compared for defining a failure as (1) failure of the instrumentation function in a single tube, and (2) failure of the instrumentation function in two adjacent tubes.

1. Single Tube Failure

A single-tube failure criterion implies that if the instrumentation function is lost in any of the channels, the reactor will be shut down and the defective instrumentation will be repaired or replaced. It is assumed that there are 91 independent identical channels and that the probability of failure is the same for all channels. The reliability for all of the channels is

$$R_{(reactor)} = R^{91}$$  \hspace{1cm} (1)$$

where $R$ is the single-tube reliability and $R_{(reactor)}$ is the composite reliability for all of the channels. The reliability as used here is defined as the probability of performing without failure the required functions under the actual in-use conditions for a period of 1 year. The relationship between the single-tube instrumentation reliability and the reactor in-channel instrumentation reliability as determined by equation (1) is listed in Table VII and is shown graphically in Figure 34 (single-tube criterion).

The required single-channel reliability can be determined if both the allowable reactor unavailability resulting from the in-channel instrumentation, and the associated lost-time-per-failure are known. A reactor unavailability of 2 hr/1000 hr is used as the basis for the reliability requirement. This number was determined by engineering judgment to be a reasonable allocation based on 75% reactor availability. The lost-time-per-failure is predicted to be 56 hr$^{(23)}$ as follows:
## TABLE VII

**REACTOR RELIABILITY VERSUS SINGLE-TUBE RELIABILITY**

**FOR TWO FAILURE CRITERIA**

<table>
<thead>
<tr>
<th>Single-Tube Reliability</th>
<th>Reactor Reliability</th>
<th>Single-Tube Criterion</th>
<th>Adjacent-Tube Criterion</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.9</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>0.91</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>0.92</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>0.93</td>
<td></td>
<td>0.001</td>
<td></td>
</tr>
<tr>
<td>0.94</td>
<td></td>
<td>0.003</td>
<td>0.533</td>
</tr>
<tr>
<td>0.95</td>
<td></td>
<td>0.01</td>
<td>0.696</td>
</tr>
<tr>
<td>0.96</td>
<td></td>
<td>0.024</td>
<td>0.837</td>
</tr>
<tr>
<td>0.97</td>
<td></td>
<td>0.063</td>
<td>0.942</td>
</tr>
<tr>
<td>0.98</td>
<td></td>
<td>0.16</td>
<td>0.988</td>
</tr>
<tr>
<td>0.985</td>
<td></td>
<td>0.253</td>
<td></td>
</tr>
<tr>
<td>0.99</td>
<td></td>
<td>0.40</td>
<td>0.999</td>
</tr>
<tr>
<td>0.993</td>
<td></td>
<td>0.527</td>
<td></td>
</tr>
<tr>
<td>0.996</td>
<td></td>
<td>0.694</td>
<td></td>
</tr>
<tr>
<td>0.997</td>
<td></td>
<td>0.761</td>
<td></td>
</tr>
<tr>
<td>0.998</td>
<td></td>
<td>0.833</td>
<td></td>
</tr>
<tr>
<td>0.999</td>
<td></td>
<td>0.913</td>
<td></td>
</tr>
<tr>
<td>1.0</td>
<td></td>
<td>1.0</td>
<td></td>
</tr>
</tbody>
</table>
FIGURE 34. Reactor Reliability Versus Single Tube Reliability for Two Failure Criteria
Initial fuel cooldown 8
Additional cooldown before handling 4
by fuel handling machine
Removal and transfer of subassembly to 8
the gas examination facility
Replace instrument components 4
Replace fuel 8
Reach reactor criticality 16
Reach full power 8

TOTAL 56

The allowable mean-time-between-failures (MTBF) can be determined from (24)

\[ A = \frac{MTBF}{MTBF + MTR} \]  

(2)

where \( A \) is availability (= 0.998), and

\( MTR \) is the mean-time-to-repair (= 56 hr).

From equation (2) the MTBF was determined as 28,000 hr, which is equivalent to a 1-yr reliability of 0.731.

From Figure 34, for the single-tube failure criterion, it can be seen that the single-channel instrumentation reliability must be approximately 0.997 in order for the corresponding overall reliability to be 0.731.

2. Adjacent Tube Failure Criterion

The adjacent-tube failure criterion implies that the reactor will not be shut down for failure of the instrumentation function in any one tube, but will be shut down for failure of the instrumentation function in two adjacent tubes. Operation can continue with single tube failures by
dropping down slightly in power and using the adjacent channel instrumentation to calculate failed channel conditions. It is assumed that the average number of adjacent tubes per channel is five.

It can be reasoned roughly that on the average, five channels will fail before two of the failed channels are adjacent. The probability that the second failed tube is not adjacent to the first failed tube is 85/90. The probability that the third one is not adjacent to either of the first two is 79/89. The joint probability that five tubes fail without being adjacent to each other is thus determined to be 0.535; hence five tube failures is used as the base for calculating adjacent-channel probabilities.

The relationship between the reliability of a single-channel instrumentation function and the probability of having five channels fail, when there are a total of 91 channels that can fail, is given by

$$ R_{(reactor)} = \sum_{K=87}^{91} R^K $$

where $R^K$, the probability that exactly $K$ of the 91 channels operate without failure for a year, is given by the binomial distribution.

$$ R^K = C^K P^K Q^{91-K} $$

where

- $P$ is the single channel reliability
- $Q$ is the single channel unreliability ($= 1-P$), and
- $C^K$ is the number of combinations of 91 things taken $K$ at a time.

$$ C^K = \frac{91!}{K!(91-K)!} $$
The results of the calculation for various values of single channel reliabilities are listed in Table VII and shown graphically in Figure 34 (Adjacent Tube Criterion).

It is assumed that on the average, three tube instrumentation functions are restored each time that a five-tube failure occurs. Some tubes will not be repaired because they will be removed at the next scheduled shutdown for routine refueling. The mean-time-to-repair (MTR) for this type of failure includes the 56 hr as for single channel repair plus 20 hr for each of the two additional channels being repaired. The total MTR thus is determined to be 96 hr. From equation (2), and still with an availability, $A$, of 0.998, the MTBF is 48,000 hr, which corresponds to a 1-yr reliability requirement of 0.833 for the in-channel instrumentation. From Figure 34, it can be seen that the corresponding single-channel instrumentation reliability is approximately 0.96.

E. Reliability Requirements for In-Channel Thermocouple Circuits

The number of installed thermocouple circuits in each of the reactor channels is determined by the safety circuit philosophy, the operation philosophy, the reactor availability requirements, and the reliability of a thermocouple circuit. The circuit includes the sensing element, the wires, the connector and anything else that can cause the circuit to fail and which requires removal of the in-channel subassembly to repair.

Since little information is available, a calculation result is presented which can be used to determine the number of thermocouple circuits to install in each channel if the following are known:

- The channel required reliability
- The number of circuits required to operate in each channel
- The reliability of a thermocouple unit.
The binomial distribution was used as follows:

$$p_i^n = f_i^n p^i q^{n-i}$$

(6)

where

- $p_i^n$ is the probability that exactly $i$ out of $n$ circuits in the channel survive for 1 yr
- $C_i^n$ is the number of combinations of $n$ things taken $i$ at a time
- $p$ is the 1-yr reliability of each circuit, and
- $q$ is the 1-yr unreliability of each circuit ($= 1 - p$).

The reliability, $p$, of each circuit is determined from the failure rate, $\lambda$ (failures per year), by

$$p = e^{-\lambda}$$

(7)

The channel reliability is determined for $n$ installed circuits of which at least $K$ are required to operate by

$$p_K^n = \sum_{i=K}^{n} p_i^n$$

(8)

Failure rates of 0.01, 0.1, and 1 failures per year were used. The numbers of both required operating circuits and installed circuits per channel were varied from one to seven to scan the range of interest. The reliabilities of the channel thermocouple function for these various conditions are listed in Table VIII.

As an example of the use of Table VIII, assume that only required instrumentation function in the channel is the thermocouple function. Assume also that the failure rate of 0.1 failures per year is applicable, and that the safety circuits require that two thermocouple circuits be operable in order for the function to be considered successful. If the single-channel failure criterion is applicable, the channel reliability
### TABLE VIII
PROBABILITY OF HAVING AT LEAST THE REQUIRED NUMBER OF CIRCUITS OPERATING FOR ONE YEAR (R^2)

<table>
<thead>
<tr>
<th>Number of Required Operating Thermocouple Circuits Per Tube (K)</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
<th>7</th>
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<tbody>
<tr>
<td>Number of Thermocouple Circuits Per Tube (N)</td>
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<tr>
<td>( \lambda = 0.01 )</td>
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<td>( \lambda = 0.1 )</td>
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<td>( \lambda = 1 )</td>
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<tbody>
<tr>
<td>1 0.99</td>
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<tr>
<td>2 0.98</td>
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<tr>
<td>3 0.97</td>
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<tr>
<td>4 0.999 0.961</td>
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<tr>
<td>5 0.999 0.951</td>
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<tr>
<td>6 0.998 0.941</td>
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<tr>
<td>7 0.998 0.932</td>
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<tr>
<td>1 0.905</td>
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<tr>
<td>2 0.991 0.819</td>
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<tr>
<td>3 0.999 0.975 0.741</td>
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<td>4 0.997 0.953 0.671</td>
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<td>5 0.993 0.926 0.607</td>
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<tr>
<td>6 0.999 0.986 0.895 0.549</td>
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<tr>
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<tr>
<td>1 0.368</td>
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<tr>
<td>2 0.600 0.135</td>
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<tr>
<td>3 0.748 0.306 0.050</td>
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<tr>
<td>4 0.763 0.768 0.144 0.018</td>
<td></td>
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<tr>
<td>5 0.896 0.602 0.264 0.064 0.007</td>
<td></td>
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</tr>
<tr>
<td>6 0.933 0.710 0.386 0.136 0.028 0.003</td>
<td></td>
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<td></td>
<td></td>
</tr>
<tr>
<td>7 0.954 0.792 0.506 0.226 0.068 0.012 0.001</td>
<td></td>
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must be 0.997 or higher (from Figure 1). It can be seen from Table II that for \( \lambda = 0.1 \) and for two circuits per tube required, there must be four installed circuits. If the adjacent-channel failure criterion is applicable, the channel reliability must be 0.96 or higher and from Table VIII it can be seen that three installed circuits satisfy this condition.
IV. SAFETY CRITERIA

Safety criteria for the FFTF instrumentation have been formulated and are undergoing intensive review as the design progresses. Criteria presented herein are current at this time but are subject to future modification and are intended to be used as part of the present bases for driver fuel instrumentation. Safeguards criteria for the FFTF, of which this section is a part, will be presented in separate documentation. Criteria, together with brief rationale, are presented below.

Instrumentation for measuring the neutron flux level in the reactor core will be provided for:

- Continuous monitoring of the shutdown flux level with fuel in the core.
- Monitoring startup flux and flux responses to controlled reactivity insertions.
- Monitoring fluxes continuously during operation from low to high power levels.
- Continuous monitoring of power operations with linear response to power for overpower factors up to at least 125%, and an overpower factor of at least 300% before saturation of the instrument.

Continuous measurement and readout of the flux level in the reactor is required to provide an indication, with a short time delay, of the reactivity status of the reactor. The reactivity status can be determined very accurately by the response of the flux level with time for all conditions except shutdown. Under shutdown conditions, special equipment is required to estimate the subcritical status, but flux monitors still will indicate when close to critical by the response to small reactivity insertions.

The instrument margin required above the normal power level provides assurance that the response at full power is still rather linear and that
the instrument has not saturated. (At saturation, the output of the instrument does not respond to an increase in flux since it is at its peak output.)

The flux monitoring instrumentation will be redundant and independent to provide continuous monitoring of the power level under all conditions. At least one-half decade overlap will be required in transferring control from one instrument to another. Individual channels will be independent to the logic network.

The functional requirements of the safety system emphasize the high reliability under which it must operate. This requires that all components be designed and manufactured to high industrial standards and that the system logic, through redundant components and fail-safe circuitry, provide high probability of correct operation even with system internal failures.

Redundancy can imply totally independent redundant systems or redundant components within that system. In the safety system it is required that system redundancy and independence be provided such that a single component or circuit failure cannot result in inoperability of the system function. At least two channels are required for each function.

Nuclear instrumentation will cover the three ranges of operation: startup or low level, intermediate level, and power or high level. The effective range of each set of instruments will cover operation within that range with at least one-half decade overlap in the transitions between ranges to give continual flux level indication and system protection.

In-core or submerged instrumentation will be structurally capable of maintaining operability under abnormal transient conditions which may result in moderate fuel damage. Subcritical flux monitors, once withdrawn after startup, will be capable of surviving an accident resulting in gross fuel failure.

Knowledge of the reactivity status of the reactor is required even under postaccident conditions up to the DBA. For that reason it is
necessary that some flux monitors be capable of surviving the transient conditions associated with the accident and provide a valid flux signal afterward. This capability may be provided by the design of the instrument or by protective structure around the instrument.

*Instrumentation will be required to verify adequate cooling flow for every fuel subassembly and control rod channel at a reactor power level below that which could cause sodium vaporization in a plugged channel.*

During shutdown periods, there may be a greater tendency for a buildup of sodium precipitates within the subassemblies due to the lower sodium temperature and lower flow. Verification of adequate flow on every subassembly and control rod channel is required before operation at a level which might result in sodium vaporization or fuel meltdown within the plugged channel.

Flow may be verified by pressure drop or a flowmeter, which might be any of a number of different types. Comparison of the outlet temperature with the bulk inlet to verify flow requires some finite power generation and also some flow to transport the heat from the core to the temperature detector if the detector is located downstream out of the core environment.

The flow detectors may be a threshold-type instrument for verification only of a minimum flow and need not give an indication of the actual flow. Temperature detectors may be capable of providing this capability.

*Instrumentation will be provided to measure outlet coolant temperatures for every fuel subassembly and control rod. Development of flowmeters suitable for use on every subassembly will be carried out; the extent to which a given type of instrument will be incorporated in the final core design will be established in further conceptual and design studies of potential conditions for loss-of-channel-flow, potential consequences of loss-of-flow, and detection and response capability of possible instrumentation.*

Outlet coolant temperature instrumentation will provide a direct measure of cooling sufficiency and an indirect indication of flow
conditions. Gradual deterioration in flow through any subassembly can be detected. Plugging of the flow passages sufficient to cause significant reduction in subassembly flow can be detected in time for corrective action unless the plugging is both rapid and extensive.

Further study is needed to assess the potential for additional protection against single subassembly flow reduction which could be provided by flowmeters.

In addition, further study of the consequences of flow blockages is needed, including improved experimental data on sodium boiling dynamics and the effects of fuel failure. The safety system will be capable of protecting against the probable reactivity effects of a flow blockage leading to fuel meltdown. If such an event would lead to local high pressures capable of causing propagation of the disturbance to other subassemblies, flowmeters on each subassembly channel could become a requirement.

**Bulk coolant conditions will be monitored to indicate:**

- **Inlet plenum bulk pressures and mixed temperatures**
- **Outlet plenum bulk pressures and mixed temperatures.**
- **Hold-down plenum bulk pressure.**

Bulk coolant conditions are required to indicate gross imbalances within the system. The inlet and outlet plenum pressures will indicate gradual overall plugging of the core by increased pressure drop for a given flow rate. The hold-down plenum pressure (if hydraulic hold-down is applied) will verify the integrity of the hold-down system and indicate any deterioration of the system by possible increased leakage during operation.
Each control element channel in the core will be instrumented redundantly for the following control indications:

- Control element full in
- Control element full out
- Loss of hydraulic hold-down or the backup hold-down feature
- Outlet coolant temperature
- Element position

Continuous monitoring of the reactivity status of the reactor is required and includes monitoring of the long-term reactivity trends. Calibrations of the control rod system will be required so that the total reactivity worth of the rods at any particular time may be evaluated from their positions in the core. These reactivity balance checks will indicate unusual trends and verify the minimum shutdown margin available in the event of some failure. The full-in, full-out indications are backup to the actual position indicators. Loss of hold-down or high outlet temperature indicates some fault in operation with the control rod and requires corrective action before occurrence of another fault or failure of the rod.

Each safety element channel will be instrumented redundantly for full-in and full-out position indications, an indication that the rod is moving in or out at its design speed when required to do so, and loss of hydraulic hold-down or its backup feature.

These requirements for the safety rods correspond to those for the control rods except that actual position indication between full-in and full-out is not required since the safety rods will not operate in intermediate positions.

The thermal-hydraulic instrumentation will be used to evaluate reactor power which will then be used for on-line computation of the core reactivity status. Comparison of this computation with the computed reactivity from control rod position indications will be used to signal out-of-limits occurrences.
Continual reactivity balances will be required to provide a signal of long-term deviations from precalculated trends considering burnup, temperature, and loading effects. Analyses of various types of system failures which may be detected in this manner will be performed to define the required sensitivity.

At least two independent channels will be provided for each function in the safety system.

Channel redundancy reduces the probability of not sensing a system error and taking action due to a failed component in the system. Redundancy also provides a means of cross-correlating signals to verify consistent operation of both channels. The usual redundancy for safety system channels in service during power operation is expected to be three, with two out of three trip logic.
V. STATUS OF INSTRUMENT TECHNOLOGY

A. Assessment of Present Instrument Technology

The present technologies for measurement of driver fuel channel temperature, flow, fuel failure, pressure and neutron flux are reviewed with respect to such characteristics as response time, sensitivity and reliability. Signal transmission and data reduction are discussed for all of these instrument systems together, rather than for each separately.

1. Temperature Measurement Technology

In this review, only subassembly coolant measurement will be covered. Fuel pin temperature measurement is not considered here as driver fuel instrumentation. Therefore, the temperature measurement range will be from 700 to 1200°F for this review. Because chromel-alumel thermocouples have been so widely used in reactors for this temperature range and because there has been relatively good experience with them, only this type will be reviewed for its characteristics. It should be noted that although there are other methods for measuring temperature, such as the resistance temperature detector and the pneumatic thermometer, the thermocouple has been a clear choice for the driver fuel subassembly application.

Selection of the thermocouple as the clear choice for temperature sensing was so firmly based in the experience of the community of reactor designers that a vigorous program to screen the available products of various vendors has been recommended. More than for other sensors types, thermocouple use is well founded in terms of feasibility of concept in high-temperature sodium, in terms of proven useful experience and for the estimated performance it can provide specifically in the FTR environment.

The pneumatic thermometer suffers from the intrinsic complications of an involved concept having working parts and a susceptibility to corrosion and line plugging.
Resistance temperature detectors suffer from the types of material stability problems that have also limited the use of strain gages in high power reactors for in-vessel surveillance. Reproducibility is degraded considerably as metal phase changes occur. Since the phenomenon of electrical resistance is so susceptible to change by volume-integrated effects of such destructive influences as high and varying temperatures and damaging radiation fields, no use is planned for RTD's except for materials testing purposes.

a. **Response Time**

For the SEFOR Reactor\(^{(26)}\) the chromel-alumel thermocouples are expected to have a time constant of 0.1 sec. This unusually low response time is to be achieved with 0.012-in. wire, MgO insulation and Inconel sheathing of 0.062-in. OD and 0.009-in. wall thickness. These are apparently grounded thermocouples to achieve this fast response. System response time which includes coolant transport time, delay due to heat capacity and necessary filtering could easily be five to ten times this figure. For the FARET Reactor\(^{(27)}\) plans had been made to use chromel-alumel wires (size not given), with alumina insulation and a 0.040-in diameter 304 stainless steel jacket. The junction points were to be brazed to the fuel subassembly hexagonal can. The response time was not given but from the description of its construction, the response time should be similar to that of the SEFOR thermocouples described above.

In EBR-II\(^{(28)}\) time constants of more than 15 sec have been measured for the chromel-alumel thermocouples used there but this high figure included the effect of the 5/16-in. tubing thermocouple well.

In Fermi\(^{(29)}\) tests were run out-of-reactor for new iron thermocouples by plunging the sheathed thermocouple into liquid sodium. Time constants of about 1 sec were obtained for grounded, and about 6 sec for ungrounded thermocouples.

In the Phenix Reactor chromel-alumel subassembly coolant thermocouples are expected to have a response time of 1 sec.
Response data on installed and operating reactor thermocouples are generally not available. In most cases it appears that the as-installed response time is unknown. However, it is no easy matter to make good uniform tests. Environmental variables such as the heat transfer and heat capacity characteristics of the material surrounding the installed thermocouple must be considered. It should be noted that the unusually low time constants (0.1 sec) mentioned for the SEFOR thermocouples refer to the time required for the installed thermocouple to respond to a step change in temperature of the coolant at the surface of the thermocouple sheath. Other time delays due to the heat capacity of the assembly to which the thermocouple may be attached, and the coolant transport time from the end of the fuel to the thermocouple location (for example: 10-ft distance at 20-ft/sec coolant velocity = 0.5 sec) must be added to the delay time of the thermocouple itself, and if it is necessary to place the thermocouple in a well, the delay time can be additionally increased by many seconds.

b. Sensitivity and Accuracy

The SEFOR coolant thermocouples (30) are expected to have a sensitivity of ± 1°F (apparently mainly due to readout equipment), and an accuracy of ± 3°F for a temperature range from 400 to 1000°F. However, this accuracy is probably not achievable in normal industrial practice and especially under FFTF conditions which will be much more severe than SEFOR conditions in terms of neutron and gamma exposure and duty cycle.

Accuracy of the Rapsodie coolant thermocouples (31) is estimated at ± 1°C. Drift in the DFR coolant thermocouples (32) after about $10^{22}$ nvt is not known exactly but is estimated to be 5 or 6°C at 600°C.

Accuracy for the EBR-II thermocouples (33) is believed to be ± 5°F for all thermocouples.

In Fermi (28), spot checks on the coolant thermocouples before installation showed that all those tested were within the ISA accuracy specifications of ± 0.75%. After installation in the reactor in 1963, an isothermal
test showed that all the thermocouples were uniform and within the same accuracy tolerance. This test was done before reactor criticality and no similar tests have been made since that time. However, it is felt that the Fermi thermocouples have not had sufficient exposure or radiation damage up to this time to appreciably affect their accuracy.

Here again, there is little exact information on the "before, during, and after" accuracies of thermocouples in actual reactor use, especially for fast reactors. There is a need to know how these characteristics change when irradiated in various combinations of fast and thermal neutron flux environments and at typical FFTF temperatures.

c. Reliability

Reliability for chromel-alumel thermocouples, when defined as a failure by shorting or breaking of the wires, is apparently good. The UKAEA found\(^{(34)}\) that their failure rate was lowered from 5\% to less than 1\% when insulated junctions instead of grounded junctions were used. Other techniques,\(^{(34)}\) such as the use of Inconel instead of 304 stainless steel to increase thermal shock resistance, controlled insulation pack density and matching of sheath and wire thermal expansions can be used to increase reliability over that known in the past. Some data to support this was obtained for these materials and others in tests for the effect of thermal shock, thermal cycling and changed insulation density.\(^{(35)}\)

The use of Inconel or perhaps lower-nickel content, nickel-based alloys or stainless-clad nickel materials as sheath material for thermocouples may well increase their overall reliability. Of course the sheath and thermocouple wire are less likely to disengage due to shock or expansion differentials if thermal expansion coefficients are nearly uniform in the couple and sheath. The designer must be able to tolerate a thickened sheath (because of corrosion contingencies) but in fact may find it useful for vibration resistance by providing a stronger component. Despite thickening, alloys such as Inconel or Incoloy 800 could furnish an improved response time due to its reduced thermal inertia relative to that of stainless steel.
In the Dounreay Fast Reactor (32) 22 coolant thermocouples were installed over a 3-yr period and 3 of these have failed. However, the temperature of the DFR coolant (700°F) is low and other conditions such as the actual life for each of these thermocouples is unknown. The very brief information available can be misleading unless other information on the installed conditions is known.

Thermocouple reliability of only 60 to 70% on 250 samples is reported (36) for fuel temperature measurements in capsule irradiations. These are grounded thermocouples, required to operate at 1800°C (3300°F). The failures (apparently open circuits) occurred mainly in regions of high thermal or neutron flux gradients. Although these are not chromel-alumel thermocouples, nor are they measuring coolant temperatures, the experience may be partly indicative of 1200°F experience in the future. Today, there is only 700-900°F experience in sodium coolant measurement in DFR, Fermi and EBR-II.

In EBR-II (28) 12 out of the 25 originally installed, metal braid-sheathed chromel-alumel thermocouples failed due to an unintended sodium leakage into the braid, and presumably into the thermocouple wells. These 12 were replaced with 1/8-in. OD thermocouples of the same type, but sheathed in solid 304 stainless steel. The original 13 have been in use since 1963 and these, together with the newer ones, are felt to be accurate and reliable. Failures, other than the 12 mentioned, have been relatively few and failed units can be replaced during a shutdown.

In Fermi (29) overall failures have been about 2 or 3% since the initial sodium filling. However, a failed thermocouple is not replaceable, mainly because the sheath is welded at the top of the vessel.

Published failure rates consistently consider only gross or definite failures such as a short or open. There is also a drift failure such as a continually increasing error, which is very difficult to identify in installed conditions unless there are redundant or nearby thermocouples for comparison. Even in this case the drift error may not be apparent if all the thermocouples were installed at the same time, in the same environment, and if all drifted in the same direction.
2. Flow Measurement Technology

In the reference document (37) Popper, Wiegand and Glass surveyed in-core flowmeters for FFTF conditions. Their conclusion was that the permanent magnet, turbine and eddy-current types of flowmeters were the most likely to be useful and somewhat in the order listed. All three would require further development and testing and all have drawbacks.

The main disadvantage of the permanent magnet flowmeter is that its sustained accuracy is uncertain due to magnetic flux instabilities at high temperature and high neutron flux conditions. The main problem of the turbine flowmeter is with bearing materials for the 1200°F sodium conditions, and if a jamming failure should occur, there could be as much as a 36-psi (37) drop (six times normal) across the flowmeter. This restriction could create a hazard for the fuel subassembly. The main disadvantage of the eddy-current flowmeter may be that its high driving current requirements (amperes) at some frequency such as 100 hertz may cause noise problems for other instrument measurements such as thermocouples. In addition, this type has unknown accuracy. Its use by the British in the PFR (38) is primarily as a rapid flow change detector rather than as an accurate flowmeter.

This discussion should not be taken to mean that these three types are the only flowmeters which can finally be used for the FFTF. Simpler or superior types may be feasible in the future. However, for the purposes of the following discussion on speed of response and other characteristics, only these three types will be considered.

a. Flowmeter Response Time

As far as the flowmeter itself is concerned, the response time of the permanent magnet and the eddy-current types are essentially instantaneous, with response time such as a few milliseconds. However, for practical purposes, it will be necessary to add delay time (estimated at about 100 msec) in the form of filtration at the amplifier or other places in
the signal transmission and readout circuits to avoid spurious trips. For the turbine type\(^{(37)}\) a typical time constant is about 8 msec for the flowmeter itself, but to this time must be added the delay of the frequency-to-dc converter. Therefore, its overall time constant would be similar (about 100 msec) to that of the other two.

b. Sensitivity

The SEFOR\(^{(26)}\) permanent magnet flowmeter gave an output of 12.5 mV at full sodium flow (72 gpm) at 700°F and the signal was found to be linear with flow rate. Accuracy of \(\pm 10\%\) between 10 and 25 gpm and \(\pm 5\%\) above 25 gpm is predicted.

The FARET, PM flowmeter\(^{(27)}\) was expected to have a sensitivity of about 0.5 mV/ft/sec for the 17-in. ID flowmeter. If the same magnetic field could be attained in the FFTF flowmeter, this would mean an output of about 15 mV, assuming a full flow velocity of 30 ft/sec. This type of flowmeter has an output of zero at zero flow. Considering these factors and experiences it is reasonable to expect a measurable output change when the flow changes by 2%. In other words, a sensitivity of at least 2% of design flow should be attainable.

It is assumed that flowmeters required for safety purposes in driven fuel ducts will not provide the capability for carrying out experimental measurements which call for high accuracy. Monitoring of decay heat rejection in terms of measuring coolant flow and temperature rise will be best accomplished on the bulk primary loop piping, for example.

It is expected that the eddy-current type's sensitivity should be similar to that of the PM type, although no data have been found for this kind of test for FFTF conditions.

The turbine flowmeter\(^{(37)}\) has a frequency output proportional to flow. If the frequency converter were reasonably sensitive, the flowmeter together with the converter should provide the same resolution as the other two types.
c. Reliability

Reliability is based on experience and there has been little of this in in-vessel flow monitoring. The SEFOR PM flowmeter (26) operated for 1200 hr in 1000°F sodium out-of-reactor before a lead failure occurred. Also, no measurable magnet field strength change was detected after 9700 hr at 700°F for the SEFOR magnets. Similar magnet stability at about 1000°F has been reported by others. Testing to 1200°F is presently being done at ANL but stability at that temperature is not yet well established.

Turbine flowmeter work at high temperatures has generally been done in pipes that are not submerged (as channel flowmeters would be in the FFTF) so the pickup coils were in a gas atmosphere at much lower temperatures. Radiation damage to coils and bearings has not been adequately tested.

For the eddy-current type of flowmeter, there is no experience data applicable to FFTF conditions. In short, there is a serious lack of real-life reliability information for in-vessel flowmeters.

3. Failed-Fuel Detection Technology

The reference document (39) indicates that two methods of failure detection, the fission gas method (with detection by electrostatic precipitation of fission products, scintillation counter, etc.) and the delayed-neutron method, have been used on sodium-cooled fast reactors. A third method, that of using a gas chromatograph to monitor the cover gas or gas separated from the coolant for gaseous fission products is possible but has not been applied to sodium-cooled fast reactors. Only the two applied methods will be reviewed for speed of response, sensitivity and reliability. Where applicable, these same characteristics of the location methods will be discussed at the same time.
a. **Response Time**

The response time of the EBR-II failure detection systems\(^{(40)}\) is about 20 sec (but depends on coolant transport time) for the delayed neutron type and 5-10 min for the electrostatic method. A comparable figure for the gas chromatograph,\(^{(39)}\) if it could be applied, is about 21 min for the sampling cycle.

To this time must be added the time required for locating the failure. This time is highly dependent on the sampling method and switching used. For example, in PFR\(^{(38)}\) this location system will require as much as 2 to 3 hr, for a full scan.

b. **Sensitivity**

For the EBR-II systems\(^{(40)}\) both are sufficiently sensitive to detect the fission products given off when a bare fuel pin is lowered into the reactor core during reactor operation. The delayed neutron system, however, apparently has difficulty in sensing a cladding failure if only a simple gas release occurs. The electrostatic system will detect both a gas leak and a fuel exposure. The suggested advantage of having both systems is that by analyzing the responses, the type of failure can be diagnosed. Since the half-life of the longest-lived delayed neutron precursor is less than one minute, the delayed neutron system is inoperable after shutdown, so though it is needed for fast response during operation, both a fission gas and a delayed neutron system are required.

At Fermi\(^{(42)}\) the electrostatic detector detected the fuel failure of October 1966. This may not, however, indicate a very high sensitivity because this was a gross failure.

c. **Reliability**

At EBR-II\(^{(33)}\) the delayed neutron system requires almost continuous maintenance attention for trip-level adjustment during reactor startups, periodic (3-month) preamplifier replacement, and for similar efforts. The
fission gas (electrostatic precipitation detection) system also requires a large amount of maintenance but apparently not to the degree of the delayed neutron system. It is reasonable to expect that redesign and use of more modern components could considerably increase the reliability of these fuel failure detection systems.

Reliability of the location method is even more uncertain at this time because there has been no experience with such a system. The greatest unreliability would probably be in the sampling lines and valving system because of line plugging, and other factors.

4. Coolant Pressure Measurement Technology

Coolant pressure measurements inside the reactor vessel may be desired at the channel inlet plenum, at the channel coolant outlet region, and possibly inside the driver channel itself. These measurements could be related to directly-connected driver fuel instrumentation.

The reference document(43) reviews two types of pressure instruments which are the most likely candidates for FFTF use. These are (1) electromecanical and (2) volumetric. A third possibility, the pressure balance type, is frequently used in fuel pin pressure measurements but will not be considered here because it does not readily provide a continuous readout.

The electromechanical type converts mechanical energy, by means of elastic deformation or displacement, to an electrical form by means of such conversion methods as strain gages, potentiometers, variable capacitance, variable reluctance and linear variable differential transformers.

The volumetric type utilizes a fluid pressure transmission line (capillary) to relay the signals to a pressure transmitter or converter where the sensed pressure can be converted to a pneumatic or electrical equivalent. For sodium pressure sensing, a separating diaphragm and a NaK filling in the transmission line has been used. The characteristics of these two types will be discussed below.
Response Time

The response time of the electromechanical type can be very rapid and in most cases can follow dynamic pressure changes. This response time will depend on several design factors such as stiffness of the diaphragm member, damping of the transducer movement and electrical damping due to required filtering and other circuit delays. A realistic estimate to cover all these factors for all types of transducer converters (strain gage, etc.) would be 100 msec.

The response time of the volumetric, or capillary tube transmission type, is very slow in comparison. A typical response time\(^{(43)}\) is 1 sec for 10 ft of capillary tubing and 5 sec for 100 ft. Thus, there is a definite response time difference between these two types and if the application requires dynamic response, the volumetric type is ruled out.

Sensitivity and Accuracy

Accuracies for the electromechanical types\(^{(43)}\) are in the area of plus or minus 1% of full range output for those transducers which have been manufactured for reactor use in the area of 1000°F. No sensitivity figures are available but sensitivities of 1% should be attainable. However, none of these electromechanical transducers have been shown to have or to sustain these accuracies or sensitivities under FFTF conditions.

For the capillary transmission type\(^{(43)}\) the accuracy is given as ± 1% of the full range up to 1500°F. No sensitivity figure is given but a similar estimate of 1% should be applicable to this transducer, including its required external pressure transmitter.

Reliability

Reliability experience for the electromechanical type on sodium-cooled reactors was not found. Both EBR-II and Fermi use a similar Taylor NaK-filled capillary transducer. In EBR-II\(^{(28)}\) the pressure instrument at the reactor vessel inlet has failed but a similar instrument at the reactor
outlet is satisfactory. In Fermi (29), the reliability has been good except that the resolution is poor because a 0-300 psi unit was applied to measure only about 40 psi. It is felt that the failures (mainly at EBR-II) have been in the capillary tubes themselves. These are very small (about 0.003-in. ID) and some phenomenon such as oxidation of the NaK (or of sodium in-leakage) slowly plugs the lines. The symptoms are a gradually increasing sluggishness and finally, full blockage and failure.

Less reliability might reasonably be expected from the electromechanical type due to in-vessel damage to magnetic steel, wire and insulation but there is no actual proof of this today.

5. **In-Core Neutron Flux Measurement Technology**

Neutron flux in the core can be measured by several methods, such as foil, wire, or ball irradiation, or miniature ion chambers, as long as thimbles are provided into the core or into the subassemblies. In making this review the assumption will be made that 1/4 to 3/8-in. ID thimbles can be installed at several points in the core, and ideally, into each subassembly. Such an installation requires the use of open test positions, however, and will not involve driver positions as such. Core flux maps for the FTR will normally be derived from fully explicit maps determined in appropriately corresponding loadings in the NPTR. It must be kept in mind in the following discussion that no neutron-sensitive ion chambers exist that can be used in the FTR core at full power or at 1200°F. Their use may be restricted to a mapping function at a small fraction of full power and at reduced temperatures.

a. **Response Time**

Irradiation of foils or wires requires counting at a later and delayed time so the measurement delay for this method can be considerable (hours or days) before useful information is available. The miniature chambers can give rapid response (in milliseconds) so that dynamic information relative to any core condition or rod positioning can be rapidly obtained. But it
must be presumed that these chambers could not be used at full power conditions due to short life from burnout effects.

b. Sensitivity

The long-term irradiation of foils or wire (for example, if they are sealed in a fuel pin) can give only long-term integrated flux information. This method is insensitive to local or time-dependent changes. For short-term irradiation (minutes) the sensitivity to dynamic flux conditions is considerably greater. The chambers will have the greatest sensitivity to dynamic flux changes within the core and will immediately show the effect of rod position changes. However, they cannot be operated for long at full power or at maximum operating temperature.

Accuracy of fixed-position in-core ion chambers in a water reactor is expected to be 5% when checked against wire irradiation, irradiated in the same thimble as the ion chamber.

c. Reliability

Little information on the reliability of the various systems is available. Irradiation of foils or wire should give little trouble unless the insertion and withdrawal machinery is unreliable. The miniature in-core chambers will gradually lose output due to burnup. However, because they will probably have to be withdrawn at full power, they would have reasonably long life. The life of a fixed-position, in-core chamber in the Dresden reactor is about nine months (representing about $10^{21}$ n/cm$^2$) due to 50% burnout of the 20% enriched uranium alloy.

Traversing miniature ion chambers can be made to last much longer than fixed-position chambers because the chamber would not be subject to burnout except during in-core traverses. However, the traversing mechanism and the cable flexing would probably reduce this reliability.
6. **Signal Transmission and Data Reduction Technology**

Transmission of electrical signals from inside the reactor vessel, through the vessel top, and then through containment walls will be a major problem for driver fuel instrumentation. For example, the coolant thermocouple signals will be required to pass through a transmission path which may finally have to include all of these elements:

- The thermocouple leads, insulation and sheath, all subject to damage from in-vessel temperature, gamma and neutron fluxes
- An in-vessel (possibly under sodium) connector which may suffer from pin corrosion, insulator deterioration and sodium leakage
- A top-of-vessel connector (such as at a nozzle) which may be subject to mechanical damage from fuel handling operations but operating in more moderate temperatures
- The connecting cable to the outside of the containment wall.

Of these, the path segments from the in-vessel transducer to the top of the vessel will be the critical portion of the total path. Other in-vessel signals emerging in such forms as pressures will be relatively easy to convert to an electrical form and then to transmit and operate on these in the same way as for those signals originating electrically. It is reasonable to expect that such signals as NaK-filled pressure lines emerging from the reactor vessel will be more reliable than those electrical signals originating within within the reactor vessel because of potential damage to magnetic steel, wire and insulation materials.

In general, the containment wall will be the nearest point at which other signal conditioning and handling equipment will be applied. For those signals which are to be used for tripping in the safety circuit, the components (for thermocouple signals) would be reference junction compensators, filters, amplifiers, trip controllers and safety circuit relays. For the remainder of the signals (and the safety signals where safety and
independence considerations permit tie-ins with data scanning) the transmission paths will be through signal conditioning components, amplifiers, multiplexers, analog-to-digital converters, the computer, and in some cases, actuators. While all of this equipment is complex as a whole, the current state of technology\(^{(45,46)}\) is capable of providing adequate signal conditioning, transmission, computing, and data scanning to meet the needs of the FFTF Reactor. Several choices are available in scanning rates and other characteristics but the final choice will often be made on the basis of cost and reliability rather than being restricted to only one technologically feasible method.

This should not be taken to mean, however, that there is no great need for care in the area of signal transmission and data handling. On the contrary, the hard-won accuracies and sensitivities obtained from the in-vessel transducers must be preserved with the best available techniques of shielding, grounding, preamplification, increase of signal-to-noise ratio, and accurate and reliable signal-handling systems.

### 3. Estimates of Improvements Due to Development Efforts

The primary goal of the sensor development programs is improvement in the life and reliability of the device in FFTF service, rather than improvement in some characteristic such as response time compared with the values usually associated with that particular type of device in customary applications. Since the state-of-the-art experience reported above seldom describes sensors that have been stressed by environments comparable to those expected in-core and near core in the FFTF, comparisons of present performance in other applications and predicted performance in the FFTF are not practical to make in a realistic, quantitative manner. The first problem faced by the developer for the FFTF is to make the sensor perform at all reasonably in the unprecedented environment and confined space. Following this, the development effort strives to approach the performance achieved in existing, less demanding applications and to provide an acceptable lifetime in the FFTF environment. The following is
a list of instrument development programs presently underway under PNL direction:

- Gas Chromatograph Fuel Failure Detection
- Delayed Neutron Fuel Failure Monitor
- Gas Disengagement for Failed Fuel Monitoring
- Pressure, DP Flow and Level Sensors for In-Reactor Sodium Service
- In-Core Sodium Flowmeters
- In-Reactor Coolant Temperature Sensors
- Reactor Environmental Effects on Signal Cables
- Signal Connector for Sodium Service

1. Temperature Measurement Technology

The most significant work in the area of sodium coolant temperature measurement is in the field of thermocouples. In the time span that is of significance to this report, the time between now and the procurement of sensors for use in the initial construction of the FFTF, no basic change is anticipated in the thermocouple art. For example, no new sensor wires are expected to be introduced and adequately proof tested. However, much will be learned about the performance of chromel-alumel couples after long exposure in a fast neutron flux. The basic accuracy of the couples will not be improved, but correction factors may be learned that can be applied as a function of integrated exposure. To date some initial phases of a planned irradiation program have been carried out to screen candidate thermocouples obtained from several vendors.

Cable and insulator programs are in progress to optimize material selection and geometries to minimize errors created in the lead wires as contrasted to those created in the junction. Improved fabrication techniques are expected to increase the reliability of grounded couples, permitting this faster responding type to be used where they may not now
be acceptable. Reliability improvement is the major goal of the FFTF thermocouple program, and significant improvement is expected over the performance of typical existing thermocouples by the time they are used in the FFTF.

2. Flow Measurement Technology

Improvements in in-core flowmeters cannot meaningfully be predicted in a quantitative manner since there is so little valid experience that can be applied to the FFTF case. Little is known of the effects of fast neutron irradiation of magnetic materials. Exposure of these materials to a fast flux spectrum is expected to begin late in this fiscal year in EBR-II as part of an active program at ANL. Stabilization of the magnetic flux from permanent magnets has recently been demonstrated at 1200°F with a loss of 80% of the flux. Long-term performance in a combined high temperature and high neutron flux environment need to be demonstrated.

In a configuration such as the FFTF where the magnets must be totally enclosed in the channel, and the stray magnetic field of each magnet will interact with that of magnets in adjacent channels, the accuracies now usually associated with permanent magnet flowmeters may not be achievable. It is suggested that no anticipated improvement be factored into safety or surveillance planning. No significant change is expected to be made in response time. Reliability can be expected to be improved with experience in fabrication methods, but the higher temperatures expected in the FFTF can reasonably be expected to offset this.

A development program for in-channel eddy-current flowmeters as a backup effort is being started at ANL. Little is known of the performance of this type of device submerged in conducting fluid channels as these will be. Since the calibration is dependent on the velocity profile of the coolant, the accuracy will depend on the validity of the simulation of the actual in-channel configuration by the calibration facility. A
facility such as the proposed Auxiliary Component Test Installation (ACTI) at the Liquid Metal Engineering Center is expected to be required in order to adequately perform the proof testing and calibration.

3. Failed Fuel Detection and Location

a. Delayed Neutron Monitoring

The major time delay in obtaining a signal from a delayed neutron monitor is the transport time delay in getting the fission products from the fuel to the detector location. The delay is dependent to a large extent upon the reactor plant layout. Twenty-second delays are generally acceptable. In view of this sample transport delay, no effort will be expended to improve the time response of the detectors and electronic circuitry over what presently exists.

The sensitivity of the detection system is a function of its ability to detect a small flux of thermal neutrons in the presence of a large gamma background. Efforts are being made to test off-the-shelf detectors beyond their design limits. Optimization of the detector geometry and sensitivity, thermal and gamma shielding will be accomplished. System reliability will be increased by the development of radiation resistant preamplifiers capable of operation, but with reduced effectiveness, when local cooling has failed.

b. Fission Gas Monitoring

The principal development effort in the FFTF program is directed at obtaining an adequate gas sample for detection and location. Means are being sought to disengage a maximum amount of fission gas in the fuel channel in which the failure occurred and reliably transport it out of the reactor through vapor traps, lines and connectors to a detector. Conceptual designs are also being formulated for individual sodium sampling lines from the outlet of each channel. From this sample, gas could be disengaged outside of the reactor when the reactor is shut down or operating,
and delayed neutrons could be observed when the reactor is operating. It is expected that a fuel failure location system will be developed that will permit the identification of the particular channel that contains the failure, and that it will function with the reactor operating or shut down. Sensitivity of fission gas detection is expected to be improved by using gas chromatography.

4. Pressure Measurement Technology

Development of pressure instrumentation will be directed toward improving the radiation and temperature tolerance, material compatibility with sodium and reliability in order to produce a unit capable of operating in the FFTF in-reactor environment. Instrument miniaturization will also be accomplished to meet design requirements.

Improvements are expected primarily in these areas while attempting to maintain the existing commercial accuracy, hysteresis, zero shift and thermal shift characteristics.

Instrument development will consist of testing candidate commercially available sensors in sodium at rated temperatures and pressures. Selected units will then be modified as required for high temperature service in a cooperative program with the vendors. Radiation tolerance will be achieved by careful material selection. The modified units will be tested after each stage of development in simulated FFTF environment of process medium and temperature. Questionable prototype materials will be tested in available radiation environments.

a. Electromechanical Transducers

These transducers are composed of a thin diaphragm interfaced with the process medium which drives a bonded or unbonded strain gage, potentiometer, variable reluctance, variable capacitance or linear variable differential transformer producing a change in electrical signal output.
It is expected that the study and testing of candidate thin section materials in sodium will produce a material which satisfies both the instrument and FFTF design requirements. In addition, the sensing element will be improved to minimize the effects of high temperature, radiation, stress corrosion, fatigue and creep. Materials used for bonding, potting, insulation or fluid dialectrics internal to the unit will be modified for temperature and radiation tolerance in FFTF environment.

Existing commercial accuracy and sensitivity will be maintained if possible but may be reduced in a tradeoff to obtain the temperature and radiation resistance.

b. **Volumetric Transducer**

This device consists of a diaphragm or bellows seal in contact with the sodium and an interconnecting capillary line leading to a remote transducer. Development will be directed toward selection of seal material followed by design of a seal, capillary tube and transducer which will reduce errors due to thermal expansion and gradients. Improvements will also be expected in response time, and in the effects of pressure transients and line plugging.

c. **Force Balance System**

This system is composed of a diaphragm or bellows in which the measured pressure on one side is balanced by a pressure introduced on the other side until electrical contacts are separated.

Development will be directed toward testing the device to assure its reliable operation to $1200^\circ F$ in sodium.

5. **Neutron Flux Monitoring**

The reliability of an in-core neutron flux mapping system is expected to be improved by a prooftesting program, but no activity is funded for the FFTF at this time. Improvements in the quality of the information on neutron energy distribution derived from activational techniques is expected as a result of other LMFBR programs.
Out-of-core neutron flux monitoring in the intermediate range is being investigated in an LMFBR program at EBR-II. It is expected that the use of mean-square voltage measurement techniques will permit the monitoring of the FTR during any anticipated combination of high gamma background and low neutron flux.

C. Summary of Instrumentation Used or Planned in Other Fast Reactors

In making this survey, only operating or planned reactors were included in order to make the survey results more applicable to the future needs of the FFTF. EBR-I instrumentation is given only limited treatment because it is out of service and its proven or improved instrumentation should appear in EBR-II. The LAMPRE Reactor was excluded because it too is out of service and because its low power (1 MW_t) and limited objectives were too different from those of the FFTF reactor.

In the following list the FARET Reactor instrumentation systems were not all chosen before project cancellation so some plans are not known. For the Russian BN-350 Reactor (1000 MW_t, 1200°F outlet) which is to be built, little has been published on the instrument systems planned. Similarly, their operating BR-5 (5 MW_t, 900°F outlet) and their under-construction BOR Reactor (60 MW_t, 1200°F outlet) have little published information on instrumentation.

Table IX has been assembled to give a quick view of the existing or planned use of those instrument systems considered necessary for FFTF driver fuel instrumentation. The Russian reactors were excluded because no published information was found on driver fuel instrumentation. Each of the three types -- flow, temperature, and fuel failure -- together with auxiliary types such as flux and pressure are discussed below.

1. Fuel Failure Detection and Location

This function is divided into two categories: detection and location. The information is from reference document (39) unless otherwise indicated.
<table>
<thead>
<tr>
<th>Reactor</th>
<th>Design Power (MW)</th>
<th>Outlet Temp. (°F)</th>
<th>Coolant</th>
<th>Fuel Failure Detection</th>
<th>Location</th>
<th>Subassembly</th>
<th>Coolant Temperature</th>
<th>Flow</th>
<th>Comments; Status</th>
</tr>
</thead>
<tbody>
<tr>
<td>EBR-II</td>
<td>62.5</td>
<td>900</td>
<td>Na</td>
<td>Yes</td>
<td>No</td>
<td>26 of 47 (Above Subassemblies)</td>
<td>No</td>
<td>In operation.</td>
<td></td>
</tr>
<tr>
<td>FERMI</td>
<td>200</td>
<td>800</td>
<td>Na</td>
<td>Yes</td>
<td>No</td>
<td>25 Percent (Above Subassemblies)</td>
<td>No</td>
<td>In operation.</td>
<td></td>
</tr>
<tr>
<td>FARLT</td>
<td>50</td>
<td>1200</td>
<td>Na</td>
<td>Yes</td>
<td>No</td>
<td>Yes (On Some Subassemblies)</td>
<td>No</td>
<td>Cancelled</td>
<td></td>
</tr>
<tr>
<td>SEFOR</td>
<td>20</td>
<td>820</td>
<td>Na</td>
<td>No</td>
<td>No</td>
<td>On 6 of 109 Subassemblies</td>
<td>For 6 of 109 Subassemblies</td>
<td>Not yet in operation.</td>
<td></td>
</tr>
<tr>
<td>DFR</td>
<td>60</td>
<td>700</td>
<td>NaK</td>
<td>Yes</td>
<td>No</td>
<td>6 TC's for 342 Elements (Bulk Coolant)</td>
<td>No</td>
<td>In operation.</td>
<td></td>
</tr>
<tr>
<td>PFR</td>
<td>600</td>
<td>1200</td>
<td>Na</td>
<td>Yes</td>
<td>Yes</td>
<td>Majority</td>
<td>On 3 Dummy Subassemblies</td>
<td>Under construction.</td>
<td></td>
</tr>
<tr>
<td>Rapsodie</td>
<td>10</td>
<td>1000</td>
<td>Na</td>
<td>Yes</td>
<td>No</td>
<td>Yes, For All</td>
<td>No</td>
<td>In operation.</td>
<td></td>
</tr>
<tr>
<td>Phenix</td>
<td>600</td>
<td>1000</td>
<td>Na</td>
<td>Yes</td>
<td>Yes (At Least Partial)</td>
<td>Yes, For All</td>
<td>No</td>
<td>Under construction.</td>
<td></td>
</tr>
</tbody>
</table>
a. Failure Detection

In the detection column it can be seen that all but SEFOR have failure detection systems. It is presumed that FARET would have had such a system because they were included in both the EBR-I and EBR-II designs (also ANL-designed).

The electrostatic precipitation detection system is applied to the sampling of cover gas in EBR-II (also EBR-I), Fermi, Rapsodie, and Phenix. The delayed neutron method is applied to coolant sampling in EBR-II, DFR, PFR, Rapsodie, and Phenix. Thus, two detection systems are applied to EBR-II, Rapsodie and Phenix. These systems are considered as complementary, and not as duplicates in that each has some superior characteristics which compensate for weaknesses in the other systems.

A third detection method, that of using a gas chromatograph on cover gas or on gas separated from coolant, has apparently not been applied to a sodium-cooled fast reactor.

Regarding the delayed neutron system at DFR, it is essentially useless at this time for detection purposes because of the use of vented and exposed fuel which provides so much coolant-contained fission products.

b. Failure Location

The location column of the table indicates that only the PFR and possibly the Phenix reactors will have a complete built-in failure location system. In the PFR, sampling pipes from each of 186 subassemblies are taken downward, then up near the top of the vessel to a 186-way mechanical selector valve to route the sample to a delayed neutron detector. Not all design details are final at this time, but the sampling piping installation is definite.

In the Phenix reactor, recent information(31) indicates a system is planned similar to that of the PFR. However, earlier information(39) indicated that only partial location (location to a group) would be used.
In Rapsodie, location will be accomplished by removing the subassemblies one by one from the reactor and heating them in argon gas to obtain emission and detection of fission gases.

2. **Subassembly Coolant Temperature**

The temperature column of the table shows that only Rapsodie and Phenix\(^{(31)}\) specify thermocouples for measurement of the coolant from each subassembly. The PFR reactor will have chromel-alumel thermocouples in most\(^{(38)}\) of the subassembly outlets. Both French reactors\(^{(31)}\) will also have their thermocouples in each outlet flow. Rapsodie provides one and Phenix will have three thermocouples per subassembly.

In DFR,\(^{(32)}\) due to the core arrangement, the six subassembly coolant thermocouples (1/8-in OD, triplex) apparently measure the temperature in a region of the bulk outlet flow rather than for each subassembly. Some of these thermocouples are used for trip purposes.

The SEFOR\(^{(30)}\) instrumented subassemblies (only six) will have two chromel-alumel thermocouples at the coolant exit to measure the mixed mean temperature, and one at the coolant inlet. Properly speaking, these special subassemblies are not fully representative of driver fuel subassemblies.

For the FARET reactor no firm information was found which would indicate that 11 of its subassembly coolant temperatures were to be monitored. Such extensive instrumentation would have been difficult with its vertical core arrangement.

In the Fermi Reactor\(^{(47)}\) about 25% of its 91 (net) core subassemblies are monitored for coolant outlet temperature by iron-constant thermocouples located in the hold-down fingers over the subassemblies.

The EBR-II Reactor\(^{(48)}\) uses chromel-alumel thermocouples to measure the temperature of about half the core subassembly exit coolants. Apparently,
the subassembly coolant flows are not sufficiently isolated to provide a nonmixed flow at each thermocouple location.

3. Subassembly Coolant Flow

The survey indicated that only three reactors -- FARET, SEFOR, and PFR -- were to have full or partial capability of measuring the subassembly coolant flow. Therefore, the discussion will be limited to these three reactors.

In FARET, the capability for subassembly flow measuring, in presumably any subassembly (but not necessarily all), was being planned. Three types of flowmeters -- the magnetic type, the turbine type, and the thermal type -- were being considered. The magnetic or the turbine types were the most likely choice but further testing was to be done on all three types.

In SEFOR, a magnetic type, with magnets encased in stainless steel, has been developed for insertion at the lower end of six core subassemblies. The advantage of this lower location is that the temperature there (700°F) is lower and therefore, the magnetic flux and sensor output will be higher than at the top of the core. But its connecting signal leads must pass upwards through the relatively high core neutron flux regions to a higher temperature region and through high thermal gradients. While this flowmeter has not seen reactor use, it has been tested out-of-reactor. In one test at 1000°F, the meter lasted for 1200 hr before one of the lead wires failed at the magnet casing due to vibration. In another test, the magnetic field was found to be stable with less than plus or minus 2% variation (the limit of measurement accuracy) when tested at 700°F for 9700 hr. Strictly speaking, this flowmeter is not typical of SEFOR's driver fuel instrumentation.

In PFR, plans are to install in-core eddy-current type flowmeters in three dummy subassemblies located at the edge of the core. These will be set to trip on a rate of change of flow, and will not be used as
accurate flowmeters. Similar flowmeters will be used on each pump to provide a rapid indication of flow changes, and will also be used for trip purposes.

4. Core Neutron Flux and Other Instrumentation

a. Core Flux Measurement

In the DFR Reactor,\(^{(49)}\) four flux-measurement positions are available. One thimble in the core center can take up to a 1/4-in. OD counter chamber. The other three locations can take chambers up to 1/2-in. OD and these are located in a control rod, the inner row of the blanket, and the outer edge of the blanket. Uranium plutonium, and \(\text{BF}_3\) chambers have been used in these openings to obtain vertical core scans.

In SEFOR,\(^{(30)}\) plans are to obtain a vertical scan of the core by using wires and foils placed inside a dummy fuel element. The sealed dummy will be irradiated, unsealed, and counted to give a measure of the integrated flux for that operating period.

In FARET,\(^{(27)}\) plans were made for a dry thimble in each instrumented fuel subassembly for wires, foils, or chambers for neutron detection.

It is presumed that the other reactors have, or will have, similar means of obtaining flux traverses but the survey did not reveal specific details for them.

b. Other Related Instrumentation

All surveyed reactors will have bulk inlet and outlet temperatures and pressure measurements, together with bulk flow rate to assist in the evaluation of driver fuel conditions. Similarly, all the reactors are similar in their use of neutron flux chambers. In general, these chambers are located outside the reactor vessel except in PFR and Phenix where the size of the primary vessel forces in-vessel locations and consequently, a greater degree of cooling. Two or three chamber ranges are used with their outputs fed to log or linear amplifiers, usually with period measurements.
APPENDIX

1. SK-3-14323  Reactor and Vessel Instrumentation System Schematic and Interfaces

2. SK-3-14194  Conceptual Layout of In-Channel Instrumentation System Components
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