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EXPOSED-FUEL CALIBRATION STUDY IN EBR-II, SECOND SERIES

by

R. R. Smith, C. B. Doe,
and E. R. Ebersole

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R. R. Smith, C. B. Doe,*
and E. R. Ebersole*

EBR-II Project

January 1970

*Idaho Facilities

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EXPOSED-FUEL CALIBRATION STUDY IN EBR-II, SECOND SERIES

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ABSTRACT

Fission-product detection in the EBR-II system is annunciated by the FGM and FERD systems and by another method based on the radiometric assay of cover-gas samples for ^{133}Xe and ^{135}Xe. The FGM continuously monitors the concentrations of short-lived rare gases (^{88}Kr, ^{89}Kr, and ^{138}Xe) in the cover gas; the FERD senses the level of delayed-neutron-emitting fission products in a bypass coolant stream. The first two systems are on line and provide a continuous record for the respective index species. The third technique periodically analyzes samples for ^{133}Xe and ^{135}Xe. Under normal operating conditions, each system registers a response proportional to the power level and to the degree of system contamination by "tramp" uranium.

Deliberate introduction of exposed-fuel material increases the signals from all three systems. Defining "tramp" background as noise, and the increased response from the exposed fuel as "signal," the signal-to-noise ratio (S/N) for all three systems should be identical and independent of reactor power.

Definitive tests, based on deliberate insertion of unclad fuel material, indicated an average S/N ratio of 3.9 for the FERD and FGM systems for a single fuel pin in the Row-5 position. Lower values of the S/N ratio, based on longer-lived index species, indicated the presence of a defective fuel element that was exchanging bond sodium intermittently with flowing primary coolant.

The ratio of power generated in a control rod in its withdrawn position to that with the rod fully inserted was 0.52; the upper limit of the effective dissolution rate of U-5 Fs fuel material in hot sodium was 2.7 mg of uranium per day per pin.

Measurement of the radiological importance of cover-gas leakage during tests showed no significant change in radiation levels. Therefore, insofar as radiological aspects are concerned, continuous operation of EBR-II with at least two defective driver-fuel elements appears to be possible.

I. INTRODUCTION

From 1967 through 1969 several fission-product releases have occurred in EBR-II. In each instance previous calibration studies of exposed fuel proved exceptionally valuable in diagnosing the nature of the release and the severity of failure. Three series of exposed-fuel tests have been conducted: (1) a survey study during the period May 21 to 25, 1965; (2) a more detailed study from May 11 to 12, 1966;[1] and (3) a definitive study on November 15 and 16, 1968. The first two tests emphasized the demonstration of sensitivities and reliabilities of the existing fission-product-monitoring systems. Unfortunately, both tests were conducted under reduced-power, reduced-flow conditions. Despite an obvious inability to extrapolate these results to full-power and full-flow conditions, the information proved so useful during subsequent fuel failures that a third, and much more definitive, series of tests was conducted. Enough changes had occurred in the methodology of techniques of fission-product detection since completion of the first two studies that additional calibration effort was warranted in the third series. More specifically, the following changes were implemented: (1) an on-line, continuous cover-gas monitor (the RCGM) was installed; (2) improvements were made in the methods used for radiometrically assaying cover-gas samples; and (3) a second independent channel was added to the fission-gas-monitor (FGM) system. Since none of these changes had been tested under simulated failure conditions, the need for additional calibrational effort was evident. Moreover, it was hoped that the deliberate release of fission products to the primary-coolant and cover-gas systems could be used to assess and to delineate practical problems concerned with the operation of the reactor plant under failed-fuel conditions.

As an interesting consequence of the tests it was shown that the deliberate incorporation of unclad fuel material in the reactor core could be used as an important tool in diagnosing the nature of the release. At the time of the tests inconclusive evidence indicated the presence of a defective fuel element that was intermittently releasing sodium. Measurements of the responses of the various fission-product-detection systems, before and after the insertion of the unclad fuel material, confirmed this conclusion.

II. MODIFIED CONTROL-ROD SUBASSEMBLY

Two completely unclad (bare) EBR-II Mark-IA fuel pins were incorporated in a standard control rod. Each was surrounded by a cluster of stainless steel dummy pins. The remaining positions in the subassembly were filled with 32 standard Mark-IA fuel elements and 27 special stainless steel dummy elements.

A. Unclad Fuel Pins

The unclad pins were joined through a stainless steel transition piece to a standard end fitting, commonly referred to as a spade. Details relating to the fuel pins, transition piece, and spade are illustrated in Fig. 1. Machining operations consisted essentially of milling two flat surfaces at the bottom of the pin, slotting the transition piece, and pinning the transition piece to the fuel pin with a 1/16-in. dowel. Matching holes were drilled in the pins and transition pieces for dowelling. To prevent inadvertent disengagement the dowel was staked at both ends.

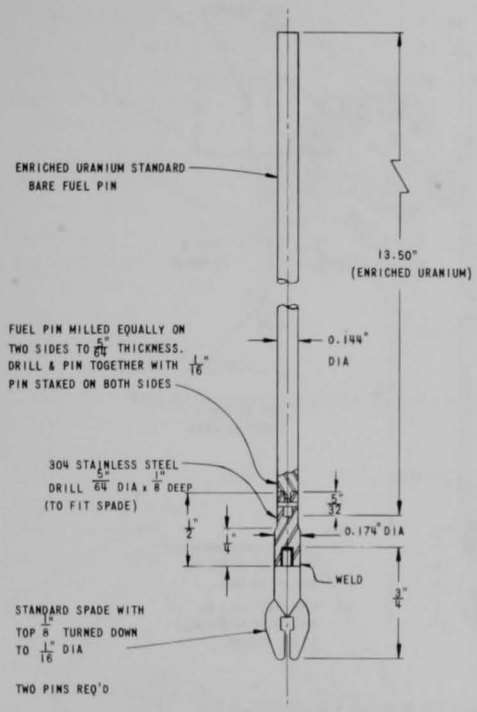

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Fig. 1. Dimensional Details of the Unclad EBR-II Mark-IA Fuel Specimens

As illustrated in Fig. 1, standard spades were modified by reducing the upper 1/8 in. to 1/16 in. The reduced sections fitted into corresponding holes formed in the bottom of the transition pieces. The space and transition pieces were heliarc welded together at the mating surfaces.

B. Stainless Steel Dummy Pins (Type 1)

Details and dimensions pertinent to the surrounding stainless steel dummy pins are given in Fig. 2. Essentially the pins consisted of 0.174-in.-dia stainless steel rods, each $17\frac{1}{2}$ in. long, fitted to standard spades in a manner similar to that described above. Spacer ribs $16\frac{1}{2}$ in. long with a 1/16-in. diameter were spot-welded vertically at 1-in. intervals along the rod. The ribs minimized radial flutter of the unclad fuel pins. The normal clearance between the unclad fuel pins and the ribs of the surrounding dummy pins was 0.004 in.

C. Stainless Steel Dummy Pins (Type 2)

Two special dummy pins, each equipped with a 1/8 x 1/8 x 1/2-in. stainless steel block welded to its top, were used to prevent the upward

movement of fuel material in the unlikely event of pin breakage. Details relating to Type 2 pins are given in Fig. 2. Both Type 2 dummies were oriented during assembly such that the restraining blocks were centered over the position occupied by the unclad fuel pins.

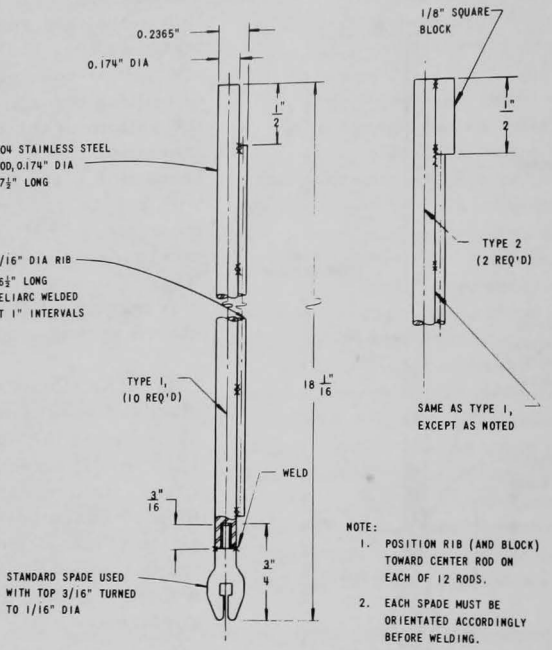

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Fig. 2. Dimensional Details of the Dummy Stainless Steel Pins

D. Stainless Steel Dummy Pins (Type 3)

A third type of stainless steel dummy pin was used to fill remaining vacancies in the control-rod subassembly. This type consisted essentially of a solid stainless steel rod equipped at the lower end with a standard spade. Superficially, Type 3 pins resembled standard driver elements; each had the same outer dimensions, and each was wrapped spirally with 0.049-in. stainless steel wire.

E. Arrangement of Unclad Fuel Pins

The locations of the two unclad fuel pins with respect to dummy pins and fuel elements are shown isometrically in Fig. 3, in horizontal cross section in Fig. 4, and in vertical cross section in Fig. 5. The 32 Mark-IA driver-fuel elements were distributed as follows: 24 in the outer row (Row 5)

with the remaining 8 located along the two flats of Row 4 in a manner which separated the test specimen from driver fuel by two rows of stainless steel dummies.

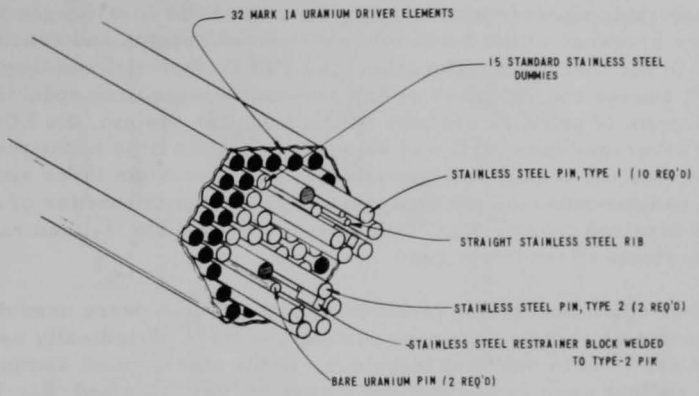

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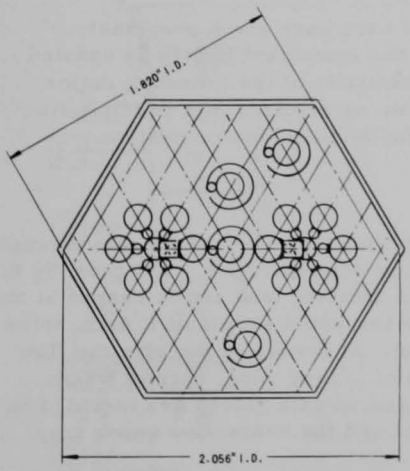
Fig. 3. Isometric Illustration Showing Locations of the Unclad Fuel Specimens



TWO BARE ENRICHED-URANIUM CENTER PINS, 0.144" DIA AND 12 SURROUNDING PINS, 0.174"-DIA, 304 STAINLESS STEEL WITH 0.0625" DIA RIBS OF 304 STAINLESS STEEL. TWO OF THE 12 HAVE A 1/8" SQUARE BLOCK WELDED TO UPPER END.
THE 14 SPECIAL PINS TO BE INSTALLED IN A STANDARD CONTROL-ROD ASSEMBLY WITH 15 STAINLESS STEEL DUMMIES WITH SPIRAL RIBS AND 32 MARK-IA URANIUM DRIVER ELEMENTS.

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Fig. 4. Horizontal Cross Section through the Modified Control-rod Subassembly, Showing Locations of the Unclad Fuel Specimens

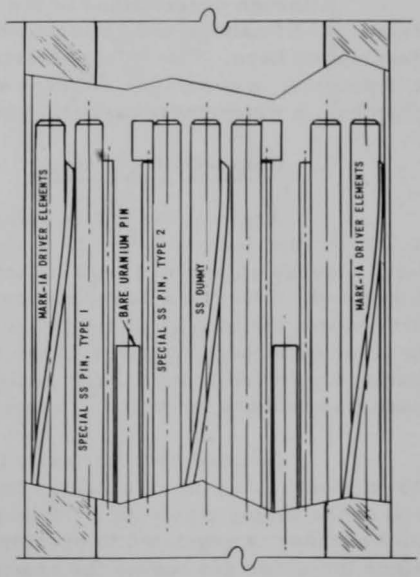

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Fig. 5. Vertical Cross Section through the Modified Control-rod Subassembly, Showing Locations of the Unclad Fuel Specimens

III. MONITORING SYSTEMS

At the time of the calibration studies two on-line, fission-product-monitoring systems were operational. One, the FGM (fission-gas monitor), senses the presence of the relatively short-lived krypton and xenon fission products in the cover gas. The other, the FERD (fuel-element-rupture detector), senses the activities of delayed-neutron-emitting species in a bypass stream of primary coolant. A third on-line system, the RCGM (reactor cover-gas monitor), was being tested at the time of the measurements but was not considered operational. In this system three single-channel, gamma-pulse-height analyzers register the intensities of specific gamma emissions from ^{133}Xe, ^{135}Xe, and unidentified short-lived rare-gas fission products in the cover gas.

Two other monitoring techniques, both off-line, were used during the tests. Small samples of argon cooling gas were periodically assayed for ^{133}Xe and ^{135}Xe in the first technique. In the other, small samples of primary coolant were radiochemically analyzed for ^{137}Cs and ^{131}I. Comprehensive descriptions of each technique are given below.

A. Fission-gas Monitor (FGM)

Although descriptions of the FGM have been given previously,[1,2] recent modifications and refinements in the equipment justify an updated description here. The FGM consists essentially of the following major components: a cover-gas delivery system, an electrostatic precipitation chamber, a water-mercury trap, and a radiation-detection system.

1. Gas-delivery System

Argon cover gas is pumped from the cover-gas plenum through a 1/2-in. pipe located in the N-2 nozzle, as seen in Fig. 6. The pipe, $9\frac{1}{2}$ ft long, extends approximately 8 in. into the primary tank and is capped at the lower end. Cover gas flows into the pipe through a series of 1/4-in. holes drilled into the pipe along the lower $4\frac{3}{8}$ in. At the top of the pipe the flow is diverted through two parallel sections of 1/2-in. pipe, each of which leads to a sodium-vapor trap. Bellows-sealed gate valves are included in each line between the vertical supply pipe and the respective vapor trap.

Located concentrically inside the vertical supply pipe is a 93-in.-long, 0.188-in.-dia Calrod heater. The heater is sealed to the top end of the supply pipe with a Conax packing gland and a lava seal. Power to the heater is regulated to provide a temperature of approximately 250°F. Since the cover gas leaves the reactor gas plenum at approximately 700°F, considerable amounts of sodium are entrained in the form of vapor. Condensation occurs in the supply pipe as the sodium vapor cools to 250°F.

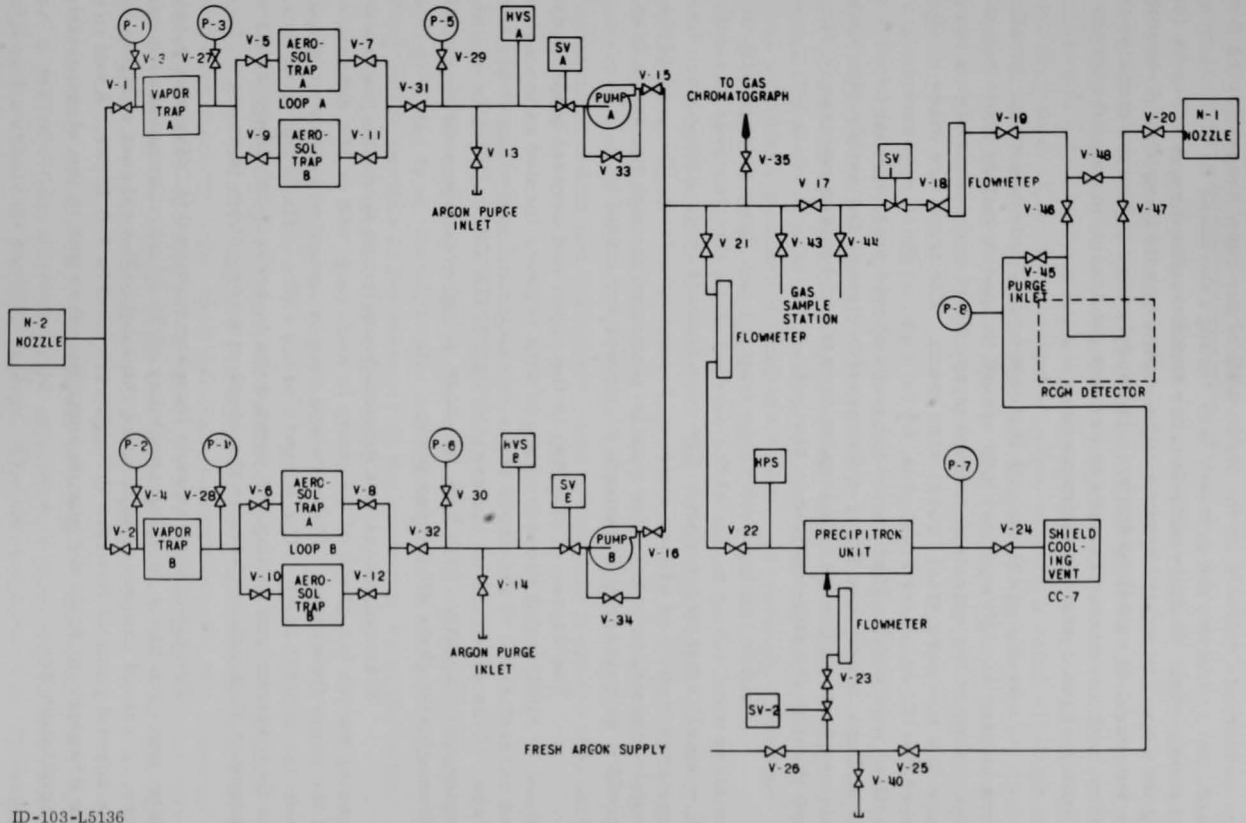

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Fig. 6. Flow Diagram of the Pumped-loop Cover-gas Delivery System

The condensate collects on the inner wall of the supply pipe, drains downward, and re-enters the primary tank through the holes in the bottom of the supply pipe. Despite considerable condensation the gas streams leaving the supply pipe still contain significant amounts of vapor. Accordingly, the two parallel exit lines between the supply pipe and vapor traps are heated with resistance heaters to prevent condensation and subsequent freezing of the entrained sodium vapor.

As the gas flows into the vapor trap (only one of the parallel lines is used at a given time) it is cooled to approximately room temperature. Most of the entrained sodium is solidified and collected in a bat of Type 304 stainless steel (York) wire mesh. The traps are made of 10-in., Schedule-10, stainless steel pipe, $6\frac{1}{2}$ in. high, of which approximately 3 in. is filled with mesh. If either of the traps should plug, special heaters melt the sodium, which then flows under gravity through the supply pipe into the primary tank. Whenever unplugging operations become necessary, the gas flow through the affected line is shut off.

After leaving each vapor trap the gas flows through 3/8-in. stainless steel tubing to one of the aerosol traps that removes essentially all residual entrained sodium. The four aerosol traps (two in each line) consist primarily of porous sintered-bronze cylinders. Because of the twofold redundancy, gas flow may be continued through the spare trap should it be necessary to remove a cylinder for cleaning.

The degree of plugging in the vapor and aerosol traps is determined by differential pressure gauges with sensors located at the entrance and exhaust sides of the vapor traps and on the exhaust sides of the aerosol traps. If the pressure on the exhaust side of the aerosol trap in use becomes more negative than 5 in. of water, a high-vacuum-sensing switch automatically shuts off the gas pump.

Solenoid valves are located downstream from the pressure-sensing switch but ahead of the pump in each loop, and on the exhaust side of the pump flowmeter. These valves may be operated locally or remotely from the control room. The purpose of the valve installation is to isolate the flow system and to shut the pump down to prevent inleakage of air or escape of radioactive gases in the event of a component leak.

The gas pumps in each loop are semisealed, diaphragm-type units, and each has a pumping capability of 46 ft^3/hr at atmospheric pressure. A valved crossover pipe from the inlet to the exhaust controls flow. The normal pumped flowrate through each loop is 4-6 ft^3/hr. A tee in the line downstream from the pumps supplies cover gas to the electrostatic precipitation unit.

2. Electrostatic Precipitation Unit

Cover gas at less than 5 psi with a flowrate of approximately 80-90 cm^3/min is pumped into the precipitation chamber and exhausted at near-atmospheric conditions to the reactor stack system. The precipitation chamber consists of two concentric cylinders: an inner aluminum cylinder with a 2-in. diameter, and an outer stainless steel cylinder with a 4-in. diameter. A 0.006-in. stainless steel wire runs axially through the inner cylinder (the anode) and is driven at a speed of approximately 18 in./min by a 3-rpm Bodine induction motor. High voltage (1500 V positive) is supplied to the anode through a conventional Teflon-insulated high-voltage fitting.

Normal flow to the precipitator is 80-90 cm^3/min. To prevent the cover gas in the precipitator from diffusing into the downstream leg of the wire, approximately 400 cm^3/min of tank argon is flushed through the upstream leg of the downstream mercury trap. The radioactive cover gas and tank argon are discharged from the precipitator through a common line to the reactor shield-cooling system.

A pressure-actuated switch and an isolation valve are located in the discharge line. The isolation valve is normally open, but if the valve should inadvertently be closed, the pressure increase would be sensed by the pressure switch which in turn would close the solenoid valve. This arrangement prevents any inadvertent loss of mercury from the open-ended mercury traps.

3. Water-filled Mercury Traps

Two mercury seals in the form of U-tubes prevent leakage of active cover gas from the wire inlet and outlet ports to the surroundings. Floating on the surface of the mercury in the upstream leg of the downstream trap is a layer of water approximately 1 in. deep.

4. Radiation-detection Systems

The radiation-detection systems consist of two independent channels. Both monitor gamma radiations emitted in the water layer of the downstream mercury trap. Both use NaI detectors located in a common cylindrical shield. Although both detectors are located at a common elevation in the shield, the access holes are oriented at an angle to ensure identical geometry between the detectors and the water fraction.

One channel (No. 1) consists of a high-voltage supply for the detector, a preamplifier, a linear pulse amplifier, a discriminator, a countrate circuit, and a countrate bucking circuit. The other channel (No. 2) consists

of a precision, single-channel, pulse-height analyzer and a countrate meter. Signals from the countrate meters are transmitted to the control room where they are displayed in countrate form on strip-chart recorders. To permit a wide variation in readout capability, in the event of a fission-product release, information from channel No. 1 is displayed in both log and linear forms, and information from channel No. 2 in log form only.

5. Principle of Discrimination against Argon-41

The major cover-gas activities pumped through the precipitation chamber include the following: ^{87}Kr, ^{88}Kr, ^{89}Kr, ^{133}Xe, ^{135}Xe, ^{138}Xe, and ^{41}A, all of which are beta active. At the instant of beta decay, each of the daughter species becomes a positively charged ion. Because of the electric field, ionized species migrate to the traveling wire and become electronically neutralized, for the wire is negatively charged with respect to the inner cylinder. The wire carries the electronically neutralized daughter species, now ^{87}Rb, ^{88}Rb, ^{89}Rb, ^{133}Cs, ^{135}Cs, ^{138}Cs, and ^{41}K, to the water layer in the downstream trap. All of these species are alkali metals and therefore react chemically with the water. The activity level of the water layer is monitored continuously with a conventional gamma-pulse-height analyzer.

Of the seven alkali metal species fixed in the water layer, only three: ^{88}Rb, ^{89}Rb, and ^{138}Cs, contribute significantly to the gross activity level. Two others, ^{41}K and ^{133}Cs, are radioactively stable; the half-lives of ^{87}Rb and ^{135}Cs are much too long (5.2×10^{10} yr and 2.6×10^{6} yr, respectively) to allow any perceptible buildup. It follows that the activities sensed by the gamma-pulse-height analyzers are exclusively those resulting from the decay of rare-gas, fission-product daughters and that the discrimination against ^{41}A is essentially complete.

B. Fuel Element Rupture Detector (FERD)

Figure 7 is an isometric illustration of the FERD system showing its physical relationship with respect to the reactor core and primary tank. The principle of the FERD system has been discussed by Porges.[3,4] However, subsequent modifications and refinements justify a more up-to-date description of the system here.

In this system a stream of molten sodium at approximately 700°F is pumped to the detection point (with a dc electromagnetic pump) at 100 gpm through a 2-in., Schedule-10 stainless steel pipe from a position near the discharge side of the primary-secondary heat exchanger.

Approximately 17-18 sec after leaving the core, the coolant flows through a 3-ft section of pipe that is completely surrounded by graphite

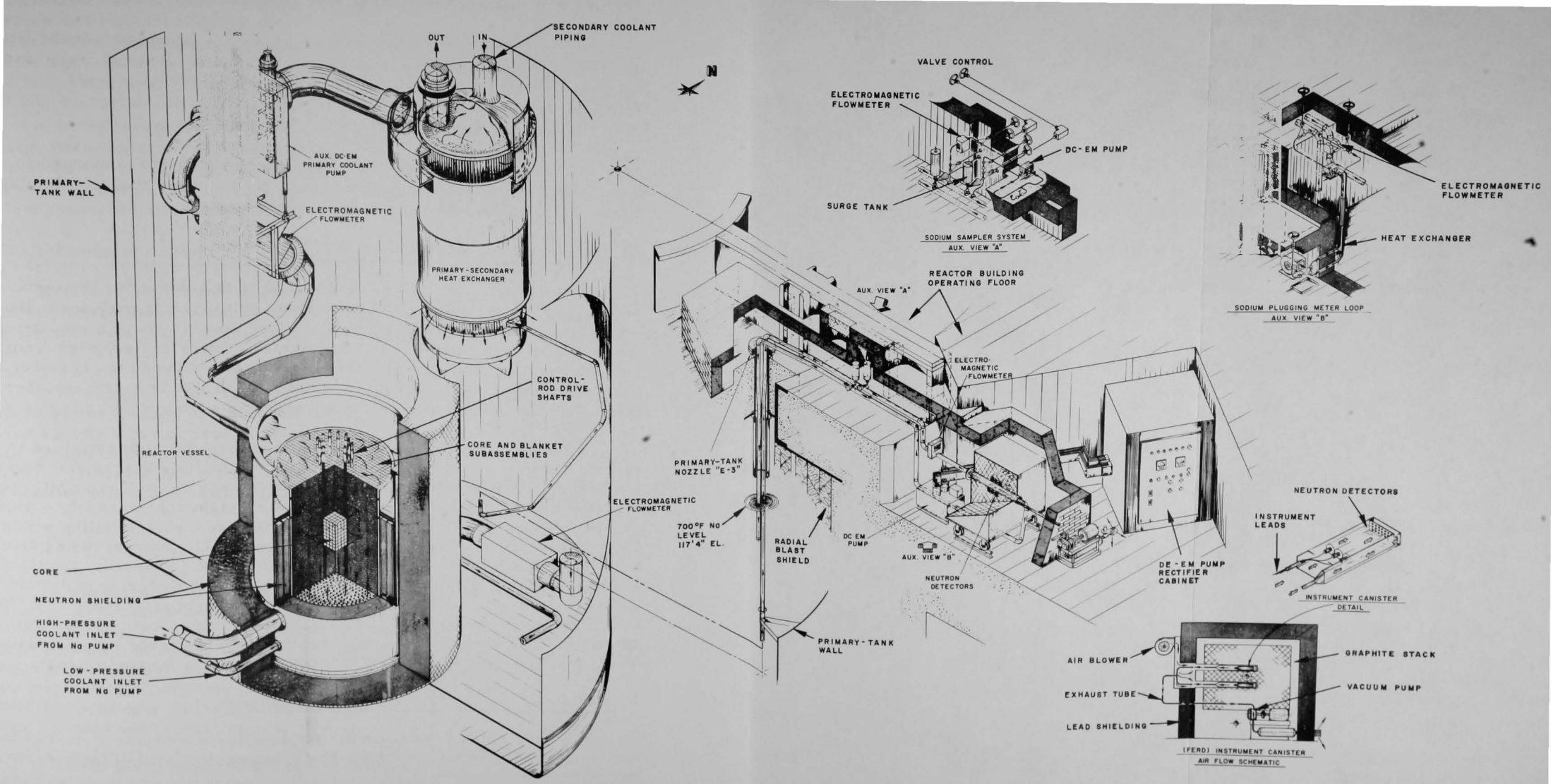

Fig. 7. Cutaway Diagram of the FERD System

moderator bricks. Located in this graphite stack are various neutron detectors that monitor the neutron level of the coolant loop.

Three independent counting channels are provided. At the time of the measurements two channels (1A and 2A) were identical; each was based on a single 2-in.-dia, 11-in.-long Reuter-Stokes BF_3 detector. The other (channel B) consisted of a gang of three 1-in.-dia BF_3 detectors and three 1-in.-dia boron-loaded chambers.*

The outputs from each of the three channels are converted to countrate form and transmitted to the control room, where they are displayed in linear form on the three-pen, strip-chart recorder.

C. Reactor Cover Gas Monitor (RCGM)

The reactor cover-gas monitor, as it existed during the tests, consisted essentially of three single-channel, pulse-height analyzers. The detector assembly for the RCGM was mounted directly above a cylindrical reservoir, 4-in. high and $2\frac{1}{2}$ in. in diameter. Cover gas supplied from the pumped gas loop (see Fig. 6) flowed continuously through the chamber at a flowrate of approximately 5 ft^3/hr. The effluent gas from the chamber was discharged directly to the reactor cover-gas plenum through the N-1 nozzle.

The detector used for the continuous monitoring of activities in the sample chamber consisted of a 2-in. integral in-line Harshaw NaI crystal-photomultiplier assembly. Amplified pulses were fed to three NMC (Nuclear Measurements Corp.) single-channel, pulse-height analyzers. The window for the analyzer monitoring ^{133}Xe was set to accept pulses falling within a 1.2-keV band embracing the 83-keV photopeak. The corresponding setting for ^{135}Xe was a band of 12 keV around the 250-keV photopeak. The third analyzer monitored pulses falling in a 40-keV band centering at 850 keV. Included in this channel were events originating primarily from ^{87}Kr, ^{88}Kr, and, to a lesser extent, from ^{138}Xe. The outputs from each of the three analyzers were converted to countrate form and displayed in logarithmic form on a three-pen strip chart recorder located in the control room.

D. Radiometric Analyses of Cover-gas Samples

Throughout the tests, cover-gas samples were taken at 30-min intervals for radiometric analysis. The samples consisted of fresh cover gas

*Subsequent improvements in detection capability included the removal of the original detectors and their replacement by six 2-in.-dia, 12-in.-long Reuter-Stokes BF_3 proportional counters ganged two per channel. Additional modifications include the installation of a second high-voltage power supply with the two outputs ganged to a common bus. To reduce the gamma flux incident on the detectors, a more recent modification (May 1969), involved the installation of a 2-in. layer of lead and a 1-in. layer of polyethylene between the coolant loop and the detectors.

contained in standard 10-cm^3 bulbs equipped at both ends with standard-taper, ground-glass stopcocks. Radiometric assays were conducted under calibrated and reproducible geometry with a precision 512-channel pulse-height analyzer (Nuclear Data, Model 180). Throughout the test series the following sources were used to check the performance of the analyzer: ^{65}Zn, ^{54}Mn, ^{137}Cs, and ^{235}U. No perceptible change in gain was noted.

Activity levels for ^{133}Xe and ^{135}Xe were established for each sample by integrating the area under the respective 81- and 250-keV photopeaks. Corrections for Compton and environmental background effects were applied with the result that activity levels cited in later sections of this report are those specifically associated with ^{133}Xe and ^{135}Xe.

E. Radiochemical Analysis of Iodine-131

Before and after the tests, samples of primary sodium were analyzed for iodine fission products. The samples consisted of approximately 10 g of primary sodium collected from the sampling station of the FERD loop.

The samples were dissolved in a methanol-water solution that contained a known amount of iodine carrier. All reduced iodine in the dissolved sample was oxidized to free iodine with $NaNO_2$ and extracted into CCl_4. Iodine in the CCl_4 fraction was then reduced to iodide with $NaHSO_3$ and back-extracted into water. The final step consisted of counting the ^{131}I activity with a standardized multichannel gamma-pulse-height analyzer.

IV. TEST OBJECTIVES

When the reactor is operated with deliberately exposed-fuel material, fission products recoil directly from the fuel surface to the primary coolant. Although the entire spectrum of fission-product species results eventually from this recoil action, only the halogens (bromine and iodine) are of immediate interest. Although krypton and xenon are used as index species, neither is directly produced to any significant extent by fission. However, both exist in the system by virtue of radioactive decay from their bromine and iodine parents.

Of the many bromine and iodine fission products produced by recoil action, at least seven are delayed-neutron emitters in the sense that beta decay to their respective rare-gas daughters results in the simultaneous emission of a neutron.

The half-lives of these species and of their respective daughters are summarized in Table I.

TABLE I. Half-lives of Pertinent Halogen/Rare-gas Fission-product Pairs

Halogen Species	Half-life, sec	Rare-gas Species	Half-life
^{87}Br	55	^{87}Kr	76 min
^{88}Br	16	^{88}Kr	2.8 hr
^{89}Br	4.5	^{89}Kr	3.2 min
^{90}Br	1.6	^{90}Kr	33 sec
^{137}I	24	^{137}Xe	4.2 min
^{138}I	6.3	^{138}Xe	17 min
^{139}I	20	^{139}Xe	41 sec

Three of the seven (^{89}Br, ^{90}Br, and ^{138}I) have half-lives too short to survive the 17-18 sec trip from their place of recoil birth, through the exit plenum and intermediate heat exchanger, then through the upstream leg of the pumped bypass loop to the detection point. All, however, decay to their respective rare-gas daughters. Some of these, in particular ^{89}Kr (3.2 min), ^{90}Kr (33 sec), ^{137}Xe (4.2 min), and ^{139}Xe (41 sec), have half-lives too short to permit the significant diffusion of these species through the primary coolant to the cover-gas plenum. In a practical sense, the four principal delayed-neutron-emitting index species for the FERD system are ^{87}Br, ^{88}Br, ^{137}I, and ^{139}I. Similarly, the index species for the FGM are ^{88}Kr, ^{89}Kr, and ^{138}Xe.

Longer-lived halogen fission products that are generated by direct fission and which are either index species or the parents of index species are ^{131}I (8.0 days), ^{133}I (21 hr), and ^{135}I (6.7 hr). Although ^{131}I decays to stable ^{131}Xe (except for a 0.7% branch to ^{131m}Xe), ^{133}I and ^{135}I decay to radioactive ^{133}Xe (5.37 days) and ^{135}Xe (9.2 hr), respectively. The latter serve as excellent long-lived and moderately long-lived index species for measurements of cover-gas activity.

A. Signal-to-Noise Ratio

Under normal operating conditions, i.e., with no sources of exposed fuel, the entire spectrum of fission products is continuously generated from fissions in fuel material which exist as an unavoidable contaminant on the surfaces of core components. The results of attempts to establish uranium impurity levels in the primary coolant are discussed in Sect. VI-D.

Because it is virtually impossible to reduce uranium contamination levels to negligible values, fission products will always be present in the primary-coolant and cover-gas samples whenever the reactor is operated. In the strictest sense the activities generated from such extraneous sources

constitute an unavoidable background commonly referred to as "tramp" or "noise." Such levels are reasonably predictable and vary little with respect to time. In some respects, the existence of a constant fission-product background is desirable, provided the background is not unreasonably high. In the first place, the relative constancy of the background provides a check on the operational condition of the various monitoring systems. In the second place, the ever-present and relatively constant background may be used as a reliable reference point in assessing the magnitude of an actual fission-product release.

To illustrate how the ever-present and unavoidable background activity level may be used as a convenient reference, the case of a given driver-fuel element is considered. Whenever the reactor is operated, fission products enter the sodium bond almost exclusively through recoil action. Because of the extremely short range of fission products in dense media, only a small fraction, approximately 0.05% of all fission products generated in the fuel material, enters the bond. Rare-gas fission products which are born in the bond from their respective halogen parents diffuse upward and accumulate in the fuel-element plenum. [A small component of the rare-gas fission product does originate from direct fission but, since the yields are small (<0.0066 and 0.19% for ^{133}Xe and ^{135}Xe, respectively), the effects of direct fission may conveniently be disregarded.] Rare-gas fission products which are generated from tramp uranium in the system diffuse upward, through the bulk sodium, into the cover-gas plenum, where they accumulate. Thus, the activity levels for a given rare-gas index species, regardless of its half-life, in both fuel-element and cover-gas plenums, follow buildup and saturation curves which have identical time dependencies. The amplitude of the buildup curve for a given rare-gas index species in the fuel-element plenum depends on the power level and the range of fission products in the fuel material. In the cover-gas plenum the amplitude depends on the power level and on the extent of the tramp-uranium contamination. Thus, whenever the reactor is operated, a certain amount of a given index species is generated from tramp sources and released to the cover-gas plenum, while a different amount is generated in the bond of a driver element and released to the fuel-element plenum. Whenever operation ceases, or is interrupted, both components decay with the same half-life. As a consequence, the ratio between the inventories of a given index species in the gas plenum of a fuel element and in the cover-gas plenum is a constant, regardless of time-power history. The same argument applies for any and all other gaseous index species. For shorter-lived index species, however, differences in the effective diffusion or migration times in the bond and in the primary sodium system must be considered. For longer-lived species, such as ^{133}Xe and ^{135}Xe, the effective diffusion time may be neglected.

The considerations outlined above provide a simple means for evaluating the maximum (hypothetical) signal expected from the release of all

gaseous fission products from a given fuel element, regardless of time-power effects. By carrying out measurements of the FGM and FERD responses and for ^{133}Xe and ^{135}Xe activities in the cover gas before and after an unclad fuel pin is deliberately inserted in the core, it is possible to establish an upper limit for the signal-to-noise (S/N) ratio. In this usage the noise is defined as the normal index activity from tramp sources, whereas the signal is defined as the increased response caused by the exposed fuel. For the exposed fuel, a specific fraction of a given halogen index species recoils directly into the flowing primary coolant. Upon decay the respective rare-gas index species accumulate in the cover-gas plenum. For an intact fuel element the same amount of the given halogen index species is generated in the fuel-element bond. The rare-gas daughter, in this case, accumulates in the gas plenum of the fuel element. If the fuel element fails and if the entire fission-product inventory of the bond and gas plenum are released, the increased response for all fission-product-monitoring systems will be the same as that noted for operating with a single exposed-fuel pin. Insofar as the responses of the various monitors are concerned, it is immaterial whether the index activity is released continuously to the flowing coolant from the surface of an unclad pin or whether the resulting rare-gas species accumulate in the fuel-element plenum and are then released to the reactor cover-gas plenum by failure. The S/N ratio will be the same for both cases, regardless of time-power history.

In a practical sense the complete release of all fission products in the bond and gas plenum of a driver element is not expected in the event of failure. All fuel elements in the core experience two pressure components: one from the static head of sodium, approximately 8 psig; and the other from the primary-coolant pumps. The latter is variable, and depends not only on axial elevation, but also upon the row in which the element is located. The pressure component associated with the primary pumps ranges from a high of 34.8 psig at the bottoms of Row-1 and Row-2 elements to a low of 7.1 psig at the top of an element in Row 6. Corresponding values for absolute pressure at these positions are 53.5 and 25.4 psia, respectively. (Atmospheric pressure at Idaho Falls is taken as 12.2 psia.) Thus, regardless of where a defect may occur, the complete voiding of the entire fission-product inventory in the bond is prevented by the combination of the static and pumped coolant pressures. As a consequence, the failure of a fuel element, followed by a partial release of its fission-product inventory, will lead to a S/N ratio smaller than that established for an unclad fuel pin located in the same position.

In a real and practical sense, the measurement of the S/N ratio for a given fission-product release may be used to eliminate driver elements as suspect. If the measured S/N ratio exceeds the value established for an unclad fuel pin, the release cannot be attributed to a single driver element. The usefulness of such information has been demonstrated by Smith et al.,

on two previous occasions.[5,6] In such comparisons care must be taken to consider the fact that measurements conducted on exposed fuel have been made for Row 5 only. Normalization of Row-5 results to other rows, including the blanket, eliminates this difficulty.

B. Power Generation in a Fully Withdrawn Control Rod

Whenever the search for a defective fuel element is narrowed to control-rod subassemblies, the search may be simplified by comparing equilibrium fission-product signals measured with a given control rod in the fully inserted and fully withdrawn positions.[7] Assuming that the defective fuel element is located in the control rod under test, the maximum fission-product signal will be generated with the control rod fully inserted. Once the intensity of the signal is established, the suspect control rod is fully withdrawn and the fission-product signal is remeasured. The difference between the two signals may then be attributed to the defective element. Obviously the difference in signals is proportional to the difference in power-generation rates with the control rod fully inserted and fully withdrawn. In planning and in analyzing the results of such tests, the relative power-generation rates must be known.

An unambiguous evaluation of the relative power-generation rates may be made from measurements of the FERD signals when the modified control rod (containing the unexposed fuel) is cycled between the fully inserted and fully withdrawn positions. In both cases the signal strengths are proportional to the area of the exposed fuel material and to the total power generated in the control rod. Since the area is constant in both cases, the ratio of the FERD signal with the control rod fully withdrawn to the signal with the control rod fully inserted constitutes a direct measurement of the relative power-generation rates. Corrections must be applied for common background effects; however, as shown in Sect. VI-B, these are small and are easily established.

C. Evaluation of Environmental Background Effects

As defined in Sect. IV-A the S/N ratio for a single unclad fuel pin is simply the ratio of the signal associated with the exposed fuel to the unavoidable ever-present tramp background. Consideration must be given, however, to other ever-present and unavoidable background components. Under normal operating conditions both the FGM and RCGM systems register small components that originate from environmental sources. For example, the detectors for both systems are located approximately 15-20 ft from the primary-sodium cold trap and the FERD system. Although all systems are well shielded, the inability to absorb completely all extraneous radiations is reflected by small environmental counting rates in both the FGM and RCGM systems. Consideration must also be given to environmental effects associated with cosmic-ray activity, airborne activity, and

the transmission of radiations through the reactor shield. As discussed in Sect. VI-C the combined effects of all environmental components are susceptible to a simple evaluation.

For similar reasons, consideration must be given to the influence of environmental effects on the registered FERD signals. In this case the primary environmental effect is concerned with the sensitivity of the BF_3 detectors to the energetic gamma radiation emitted by ^{24}Na. The evaluation of this component is discussed below (Sect. VI-A2).

D. Reaction between Unclad Fuel Material and Hot Sodium

The deliberate incorporation of unclad fuel material in the reactor core provides an excellent opportunity to study the reaction (if any) between unclad U-5 Fs fuel material and the hot (700-850°F) primary sodium. Little reaction was expected, at least for the relatively short residence time of the unclad fuel material in the core. Nevertheless, the results of chemical analyses conducted for uranium in sodium samples collected after the insertion of the exposed fuel were expected to yield upper limits for the amount of fuel material "dissolved" or dispersed in the primary sodium. A summary of such results, along with those obtained in earlier exposed-fuel studies, is given in Sect. VI-D.

E. Leakage of Gaseous Activities to the Reactor Building

Under normal operating conditions a small unspecified amount of cover gas leaks into the reactor building through various penetrations in the primary tank.

Since rare-gas fission products are released along with the cover gas, an ever-present background of rare-gas fission products (and their daughters) will exist in the reactor building. Dilution of the various activities with the air in the reactor building reduces the specific activity level to a factor of approximately four below the maximum permissible level.

Whenever fission products are released to the primary coolant and cover-gas systems as a consequence of a cladding defect, some additional leakage of rare-gas fission products into the reactor building must be expected. A quantitative evaluation of the associated radiological hazards may, in principle, be made by conducting careful building surveys during the exposed-fuel tests. The results of such measurements are summarized in Sect. VI-E.

V. APPLICATION OF FISSION-PRODUCT MONITORING INFORMATION TO FAILURE DIAGNOSTICS

Because the FERD system is sensitive to the presence of delayed-neutron emitters in the primary coolant, its response (or lack thereof) during a fission-product release provides valuable information relevant to the type and size of the cladding defect. For example, in a gas-type release from either a driver element or an experimental irradiation capsule, rare-gas fission products will be released to the flowing coolant and will ultimately diffuse or migrate upward into the argon cover-gas plenum. Eventually, the increased activity levels of rare-gas species will be indicated by the FGM and RCGM. Delayed-neutron-emitting species, presumably bromine and iodine, will remain chemically fixed in the bond which, for this type of defect, remains relatively undisturbed. Under these conditions, the FERD system will fail to annunciate a simple gas-type fission-product release.

The situation in which a defect occurs near the fuel is more complex. Since the sodium bond in the fuel element may be under positive pressure with respect to the primary-coolant system, bond sodium may be extruded through the defect. Halogen fission products, released by recoil action from the surface of the fuel, will be chemically fixed in the bond and will enter the primary sodium with the extruding sodium. If the rate of bond release is relatively rapid, i.e., if the bond contained within the cooling annulus is released over a period of seconds or minutes, the concentration of delayed-neutron-emitting species in the exit coolant stream may be high enough to be sensed by the FERD system. Such a release would eventually be sensed by the FGM because daughters of the principal delayed-neutron emitters are the index species for the FGM. However, if the release of bond sodium in the annulus occurs over a period of hours or days, the time required for a given delayed-neutron-emitting nuclide to migrate from its place of recoil birth and through the bond to the defect may be long compared to its average lifetime. Under these circumstances the leak would not be annunciated by the FERD system and could be missed even by the FGM. For a very slow bond release, longer-lived iodine fission products, such as ^{133}I (21 hr) and ^{135}I (6.7 hr), survive long enough to diffuse throughout the bond and extrude through the defect. Even though ^{133}Xe and ^{135}Xe are not directly released, the activity levels of these species in the cover gas will increase during and after bond extrusion because of their growth from their respective iodine parents. It follows that radiochemical analyses for iodine fission products in the primary coolant may be used also to indicate the existence of a bond leak that is too small for annunciation by the FERD and FGM systems.

VI. EXPERIMENTAL PROCEDURES AND RESULTS

Fundamental to the evaluation of the S/N ratio for any given index species is a knowledge of the activity levels of the index species before and after the introduction of exposed fuel. In the ideal experiment, the reactor would be run without exposed fuel long enough for all index species to reach their equilibrium levels. Then, having established the noise levels, the exposed fuel would be introduced and the reactor would again be run for an equivalent period of time. The difference between the gross signal and the noise signal would give the signal originating from the exposed fuel. Division of the exposed fuel signal by the noise signal would give the S/N ratio.

Although some attempt was made to operate the reactor under 50-MWt conditions for an extended period before the introduction of exposed fuel, the actual operating time was still insufficient to affect equilibrium conditions for the longer-lived index species such as ^{133}Xe and ^{131}I. Even more restrictive was the period of time, approximately 48 hr, allotted for completion of the exposed-fuel portion of the experiment. Accordingly, activity levels measured before and after the introduction of exposed fuel had to be corrected for the effects of differences in the time-power history.

Figure 8 summarizes the time-power history before and during the exposed-fuel portion of the tests. The two operating segments from Oct. 26 to Nov. 3 and from Nov. 5 to 14 contributed most to the generation of long-lived index species associated with tramp effects. Nevertheless, a small

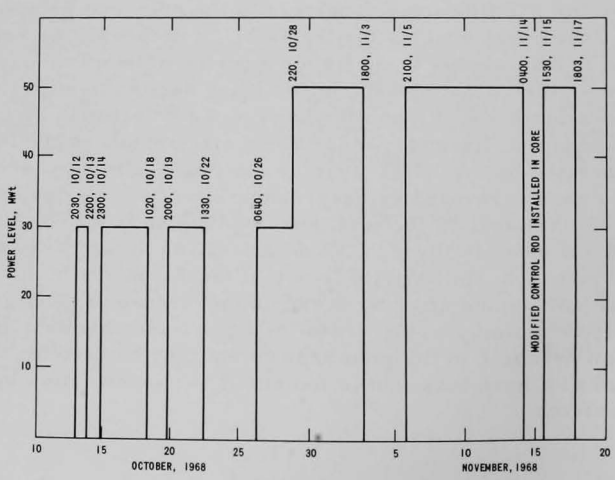

Fig. 8. Time-Power History before and during the Tests

portion was generated before Oct. 26 and, in the results cited below, the effective time-power history has been defined as the period of time between Oct. 12 and Nov. 17, during which the tests were completed.

Baseline information for the responses of all monitoring systems and for radiation levels in various portions of the reactor plant was established throughout the period Nov. 8-14. With this information established, the reactor was shut down at 0400, Nov. 14, and the modified control rod--containing the two unclad fuel pins--was installed in the core. To reduce the activity levels for ^{133}Xe and ^{135}Xe in the cover-gas plenum, gas-purging operations were begun shortly after the shutdown and were continued for approximately 36 hr. Criticality, with the modified control rod fully inserted, was reached at 1005, Nov. 15. The increase to operating power proceeded according to the following schedule: 10 MWt at 1155; 20 MWt at 1228; 30 MWt at 1300; 40 MWt at 1450; and 50 MWt at 1530. Operations at 50 MWt continued without interruption until completion of the exposed-fuel portion of the test. The final shutdown from 50-MWt operation began at 1803, Nov. 17.

A. Measurements of the S/N Ratios for the Various Monitoring Systems

1. FGM System

While baseline information was being generated with no exposed fuel in the core, from Nov. 8 to 14, measurements of the environmental background for both FGM channels were made. The wire-drive motor was turned off and the water fraction was removed with a pipet. Fresh water was added to rinse the walls of the U-tube to reduce to a minimum any residual RbOH and CsOH. Rinsing operations continued until the counting rates for both channels indicated no additional decrease. Three rinse cycles were completed.

The results of counting-rate measurements, made after the final rinse and with the wire drive off, are summarized in Fig. 9 for channel 1 and in Fig. 10 for channel 2. For channel 1, an average value of 1.0 counts/sec was estimated. For channel 2, the corresponding value was 3.5 counts/sec.

Figure 11 shows a reconstruction of the FGM strip-chart record during the exposed-fuel portion of the test. The equilibrium counting rate for channel 1 was 82.5 counts/sec; whereas that for channel 2 was 250 counts/sec.

The S/N ratio for channel 1 was established in the following manner:

First, the gross equilibrium counting rate was found to be 82.5 counts/sec. Of this, 1.0 count/sec originated from environmental

effects and 9.0 counts/sec originated from tramp-uranium sources. The signal from the exposed fuel is then 82.5 minus 10.0, or 72.5 counts/sec. Division of this value (the true equilibrium signal) by the value of 9.0 counts/sec (the equilibrium tramp background) gives a value of 8.05 for S/N ratio. Because two unclad fuel pins were used, the single-pin S/N ratio was approximately half of this value, or 4.03.

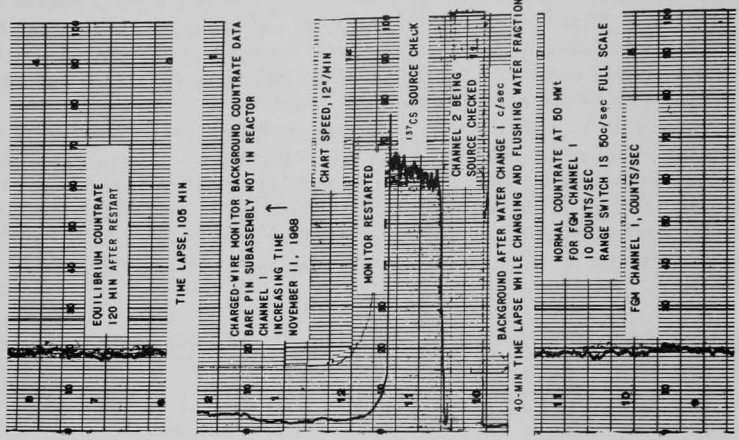

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Fig. 9. Selected Portions of the FGM Record (channel 1) before Installation of the Modified Control Rod

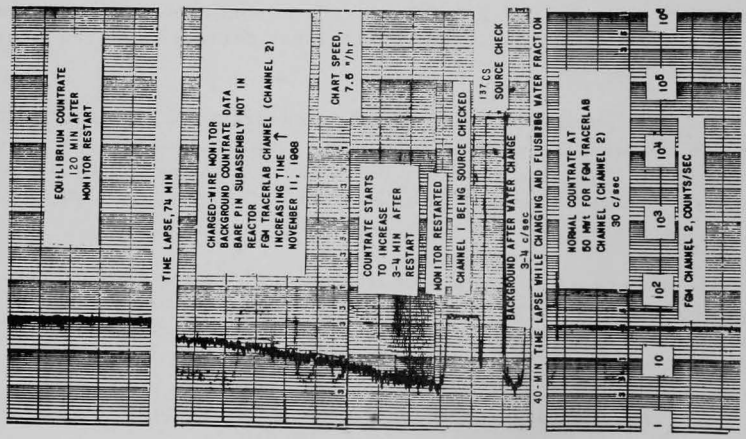

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Fig. 10. Selected Portions of the FGM Record (channel 2) before Installation of the Modified Control Rod

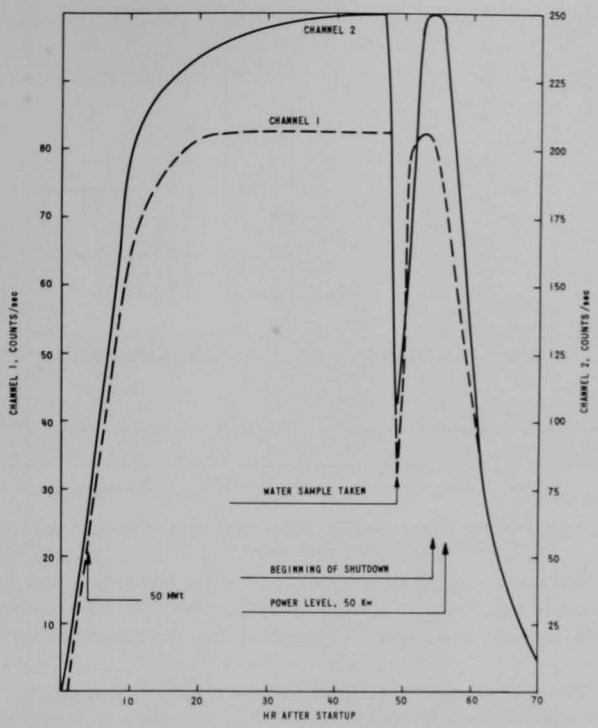

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Fig. 11. Reconstruction of the FGM Records (channels 1 and 2) with the Modified Control Rod

For channel 2 the gross equilibrium counting rate with exposed fuel was 250 counts/sec (see Fig. 11). Of this, a portion originates from tramp-uranium sources and a portion from environmental effects. Figure 10 confirms that the environmental component is 3.5 counts/sec and that the tramp component is 30.0 - 3.5 = 26.5 counts/sec. The true equilibrium signal from the exposed fuel is then 250 - 30 = 220 counts/sec. This value, divided by the tramp-uranium counting rate of 26.5 counts/sec, gives values of 8.30 and 4.15 for the two-pin and single-pin S/N ratios, respectively.

2. FERD System

Baseline values of the counting rates for each FERD channel were established on Nov. 11, i.e., before the modified control rod was installed. Figure 12 summarizes the results of these measurements. The following average values were established: for channel 1A, 17 counts/sec; for channel B, 9.5 counts/sec; for channel 2A, 5 counts/sec.

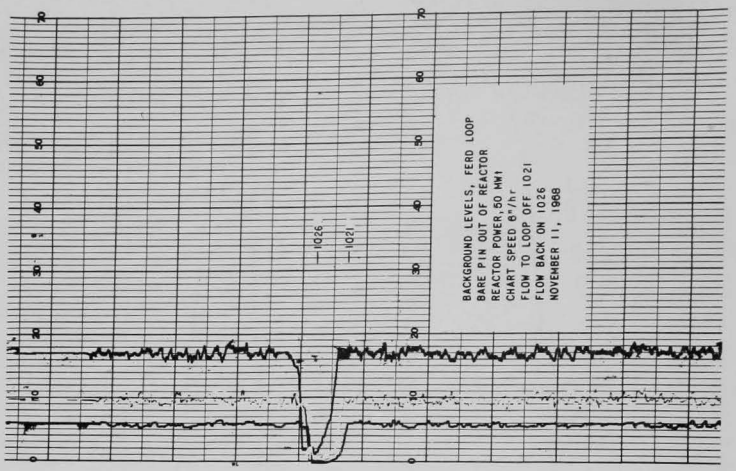

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Fig. 12. FERD Record (all three channels) before Installation of the Modified Control Rod

With the modified control rod fully inserted, the following values were established (see Fig. 13): for channel 1A, 124 counts/sec; for channel B, 76 counts/sec; and for channel 2A, 46 counts/sec.

The counting-rate component originating from environmental sources was established by turning off the loop flow and noting the counting rate after all delayed-neutron emitters at the sample point had decayed. From Fig. 13, which gives the counting rates before and after the loop flow was turned off, it may be seen that the environmental background is negligible. Accordingly, the S/N ratio for any given channel is simply the signal from the exposed fuel divided by the tramp-uranium background. The following S/N ratios were established: for channel 1A, 107/17 or 6.3; for channel B, 665/9.5 or 7.0; for channel 2A, 41/5 or 8.2. The average S/N ratio for the three channels was simply 7.2 for two unclad fuel pins, or 3.6 for a single pin.

3. RCGM System

Although it was not regarded as operational during the tests, the RCGM provided useful information. An effective equilibrium activity level of 23,000 counts/min was established for ^{135}Xe from Nov. 9-11. Normal radioactive decay and gas-purging operations reduced this level to approximately 7000 counts/min immediately before the tests. At the end of the tests the total counting rate for ^{135}Xe increased to 80,000 counts/min. Of this, approximately 21,800 counts/min originated from tramp sources during the tests. The counting-rate component from ^{135}Xe in the system

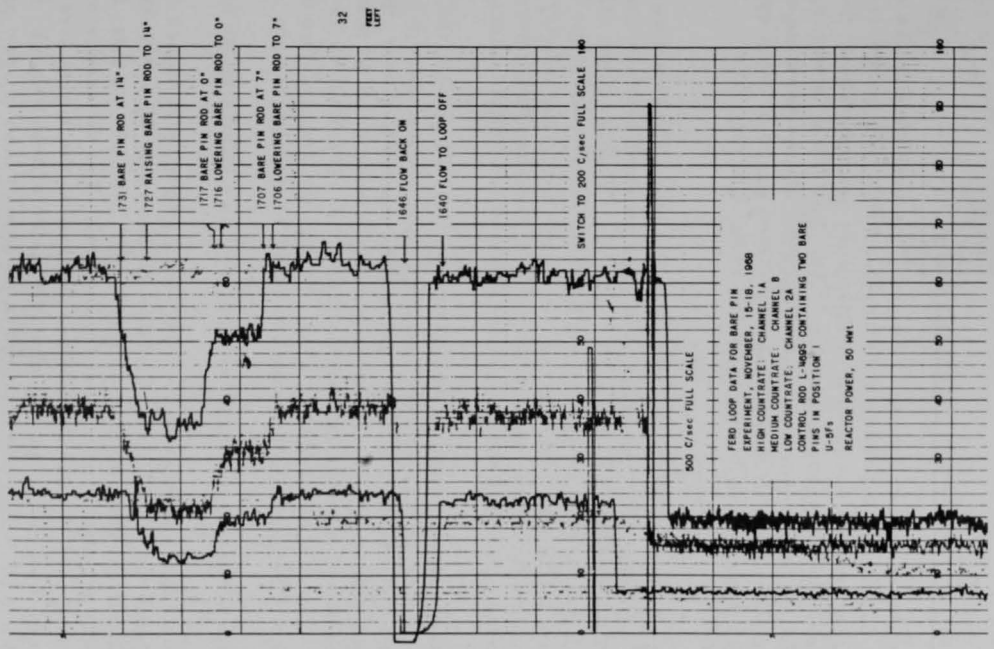

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Fig. 13. FERD Record (all three channels) with the Modified Control Rod Fully Inserted in the Core

immediately before the tests, 7000 counts/min, was assumed to be negligible at the conclusion of the tests. The signal originating from the two unclad fuel pins was then 80,000 minus 21,800 or 58,200 counts/min. Division of this value by the tramp component gives S/N ratios of 2.68 and 1.34 for two and single unclad fuel pins, respectively.

4. Radiometric Analyses of Xenon-135

Figure 14 is a record of ^{133}Xe and ^{135}Xe activities in the cover-gas before and during the exposed-fuel studies. From this record it is clear that the reactor was operated long enough in the period Nov. 6-14 to ensure equilibrium activity conditions for ^{135}Xe. For ^{133}Xe, however, equilibrium activity conditions were never achieved during the baseline experiments. Accordingly, the S/N ratio for longer-lived gaseous index species in the cover-gas system will be based on the results of the ^{135}Xe measurements.

As of 1200, Nov. 11, the ^{135}Xe activity was in radioactive equilibrium with its parent, ^{135}I. An average of experimental activities over the preceding 24-hr period was 8.6×10^{-3} μCi/ml. In subsequent calculations this value was used to define the equilibrium activity level for ^{135}Xe originating from all background sources. Although the ^{135}Xe activity was also in equilibrium during the period Nov. 12-14, fuel-handling operations during this period compromised the ability to establish reliable data.

Following gas-purging operations on Nov. 14 and 15, the reactor was started. Criticality was achieved at 1005, Nov. 15, and the operating level of 50 MWt was reached at 1530. During the ascent to power, the modified control rod was fully inserted in the core. Operations at 50 MWt continued without interruption for approximately 53 hr. The final shutdown began at 1803, Nov. 17, and ended at 1953. During the approximate 53 hr of operation the ^{135}Xe activity increased to 94.5% of its equilibrium value.

The S/N ratio for the ^{135}Xe activity was established as follows:

The experimental value of the activity level just before final shutdown was 29.6×10^{-3} μCi/ml. Division of this value by 0.945, the fraction of saturation, gave a value of 31.4×10^{-3} μCi/ml. Of this, 8.6×10^{-3} μCi/ml was attributed to normal background. The difference between these values, 22.8×10^{-3} μCi/ml, was assumed to be the signal strength originating from the two unclad fuel pins. Division of this value by the background (8.6×10^{-3} μCi/ml) resulted in a value of 2.66 for two pins, or 1.33 for a single pin.

5. Radiochemical Analyses of Iodine-131

Samples of primary sodium, taken before and after the exposed-fuel portion of the run, were analyzed for ^{131}I. Table II shows the results.

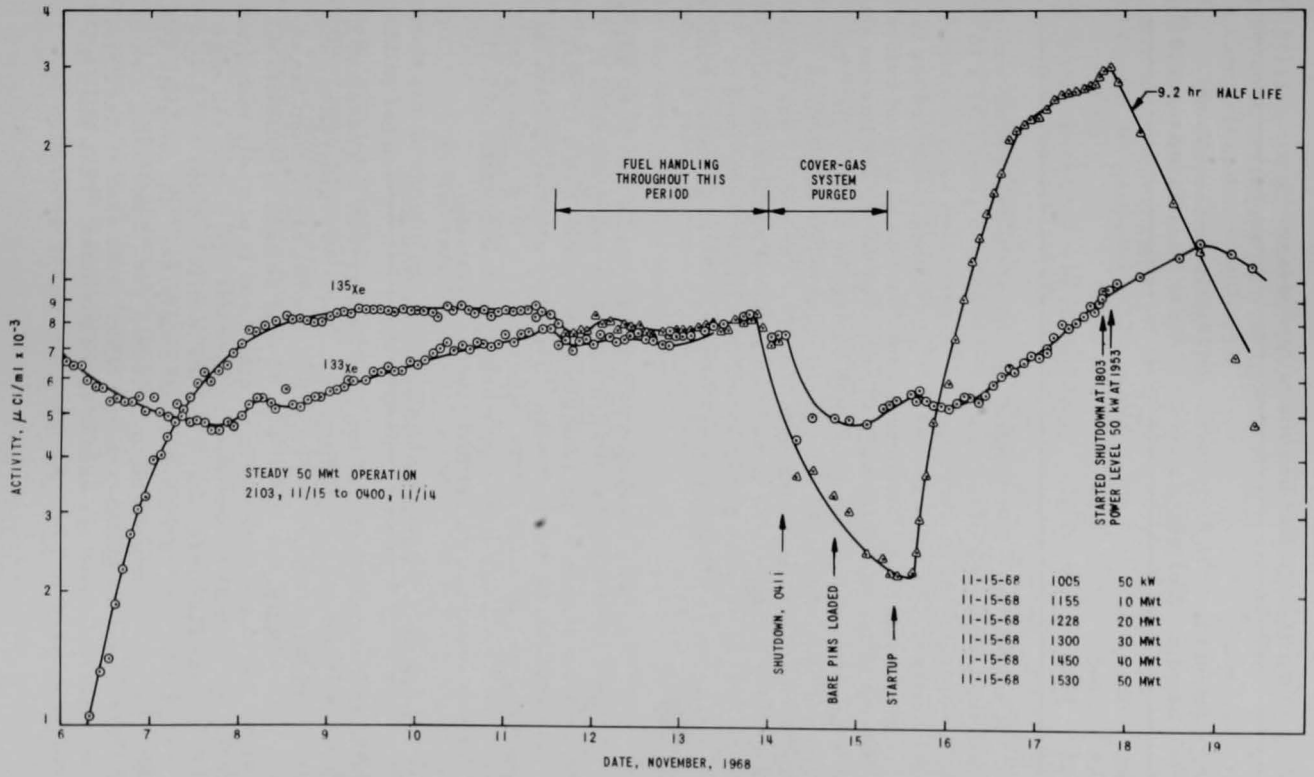

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Fig. 14. Activity Levels for ^{133}Xe and ^{135}Xe in the Cover-gas Plenum before and during Operation with the Modified Control Rod in the Core

TABLE II. Radiochemical Analyses of ^{131}I

Date of Sodium Sample	Time of Sodium Sample	^{131}I Activity dis/min-g x 10^{-3}	Fraction Equilibrium Value	Equilibrium Value dis/min-g x 10^{-3}
11/13/68	1625	1.30	0.56	2.32
11/14/68	1110	1.50	0.74	2.03
11/17/68	1900	2.60	-	-

Averaging the radiochemical results of the two sodium samples taken before the installation of the modified control rod gives 2.17 x 10^3 dis/min-g for the equilibrium concentration of ^{131}I generated from combined background sources. The value of 2.60 x 10^3 dis/min-g measured for the Nov. 17 sample includes two components: one that was generated before the exposed-fuel operation and which partially decayed during the interim shutdown and tests, and one that was generated during the actual tests. The first component was established by applying appropriate corrections for buildup and decay to the value measured Nov. 13. The second was established by applying buildup corrections to an assumed equilibrium value of 2.17 x 10^3 dis/min-g, thereby giving a combined tramp background of 1.35 x 10^3 dis/min-g as of 1900, Nov. 17 (the time the final sodium sample was taken).

The ^{131}I activity generated during the tests is then (2.60 minus 1.35) 10^3 or 1.25 x 10^3 dis/min-g. Use of this value and application of buildup corrections for the 53-hr radiation period gives 7.18 x 10^3 dis/min-g for the equilibrium signal for the ^{131}I activity originating from the two unclad fuel pins. Division of this value by 2.17 x 10^3 dis/min-g for the equilibrium ^{131}I activity from tramp origin gives S/N ratios of 3.31 and 1.65 for two and single unclad fuel pins, respectively.

B. Power Generated in a Partially Withdrawn Control Rod

If the environmental effects for the FERD system are assumed to be negligible (as the data of Fig. 13 indicate), the power-generation rate in the two unclad fuel elements is proportional to the signal from the exposed fuel. Accordingly, the data of Fig. 13 may be used to estimate the fraction of full power generated when a control rod is partially withdrawn. For example, for channel 1A the counting rate was 124 counts/sec when the control rod was fully inserted. When the control rod was withdrawn to 7 in., the counting rate decreased to 102 counts/sec. On complete withdrawal the counting rate decreased to 72 counts/sec. Subtracting a value of 9.5 counts/sec (for the tramp-uranium effect) from each of these gives the true exposed-fuel signal for each of these positions. Thus when the control rod is lowered to 7 in., the fractional power generation is

92.5/114.5 = 0.81. Similarly, for complete withdrawal the fractional power generation decreases to 62.5/114.5 = 0.55.

Corresponding values for channels B and 2A were shown to be 0.79 and 0.52, and 0.83 and 0.48, respectively. Average values for all three channels (including 1A) at the 7-in. and fully withdrawn positions are then 0.79 and 0.52, respectively.

C. Environmental Background Effects

As discussed in Sect. IV-C, one of the test objectives concerned experimental measurements of environmental background levels.

1. FGM System

During routine reactor operation the information from both FGM channels is affected by extraneous radiations. Most important are gamma radiations reaching the detector tubes from the following sources: primary cold trap, FERD loop, reactor shield, and the electrostatic precipitation chamber. As discussed in Sect. VI-A.1 and illustrated in Figs. 9 and 10, measurements of the sum of the above components were made by turning off the wire drive, flushing the water section of the water-mercury trap, and recording the instrument responses of channels 1 and 2. Values for the environmental background of channels 1 and 2 were approximately 1.0 and 3.5 counts/sec, respectively. With no known source of exposed fuel in the core, the normal equilibrium (tramp) backgrounds for channels 1 and 2 were, at the time of measurements, approximately 9 and 29 counts/sec, respectively.

2. FERD System

As illustrated in Figs. 12 and 13, environmental background effects for all three FERD channels may be established by observing the counting rates after the loop flow has been turned off for approximately 5 min (with the reactor running). During this time all delayed-neutron-emitting species in the delivery pipe have decayed; the detectors sense only stray neutrons (from the reactor shield) and a component associated with the gamma sensitivity of the detectors.

On two occasions, Nov. 11 and 15, attempts were made to measure the environmental backgrounds of all three channels. Within approximately 4-5 min after interrupting the loop flow, all counting rates in both cases decreased essentially to zero. The small residual counting rate indicated for one of the channels in Fig. 12 reflects a slight shift in the zero setting for that particular channel.

3. RCGM System

The environmental background for the ^{135}Xe channel of the FGM system was estimated roughly on Nov. 11 by removing the photomultiplier-tube assembly from its position above the source volume and placing it in a nearby shield that closely duplicated the normal shielding. The counting rate decreased from 23,000 to 1150 counts/min. Accordingly, the environmental background component associated with the indicated RCGM ^{135}Xe signal amounted to approximately 5%.

4. Radiometric Analyses of ^{135}Xe in Cover-gas Samples

As discussed in Sect. III-D activity levels for ^{133}Xe and ^{135}Xe in the reactor cover-gas plenum were established by counting 10-ml cover-gas samples under known, reproducible geometry conditions with a calibrated multichannel gamma-pulse-height analyzer. Environmental background effects were assumed to be those registered by the equipment with no gas sample in place.

D. Reaction between Unclad Fuel Material and Hot Sodium

Although sodium samples were taken before and after the installation of the modified control rod in the core (see Sect. VI-A.5), none was analyzed for uranium. During the exposed-fuel studies, changes were being made in the procedures used for determining the uranium content of the primary coolant. Nevertheless, analyses were conducted that would have reflected the loss, if any, of uranium from the exposed-fuel material to the coolant.

Following the installation of the purification-loop sampling system during the last week in Nov 1968, a 10-g sample of primary sodium was transferred to the analytical laboratory under an inert atmosphere and was vacuum-distilled to remove sodium. The residue from the distillation was dissolved in transistor-grade nitric acid and divided into two equal portions. One portion was "spiked" with 25 ng of uranium. Acid-deficient $Al(NO_3)_3$ solution was added to each aliquot, and the uranium was extracted into hexone. The hexone extracts were evaporated to dryness in platinum cups, and the residues were fused with 2% LiF-NaF.

Standard uranium samples were pipetted directly into platinum cups and fused with 2% LiF-NaF. Fluorescence measurements were made with the original sample, the sample spiked with 25 ng of uranium, and the standards. No detectable quantity of uranium was found in the sample. However, 25 ng were found in the spiked sample. From the uranium content of the standard containing the smallest detectable quantity, it was shown that the limit of detectablility was approximately 0.2 ppb. Accordingly, any loss of material from the unclad fuel pins to the coolant would

be less than the amount required to give a concentration of 0.2 ppb in the primary sodium inventory (86,000 gal). Since a concentration of 0.2 ppb corresponds to a total dissolved uranium content of 60 mg, and since the exposed fuel material was in contact with the hot primary coolant for 11 days (in the core and in the basket), the upper limit to the effective dissolution rate is 2.7 mg of uranium per pin per day (assuming, of course, that any fuel material lost from the pins becomes uniformly dispersed throughout the primary sodium).

Since the above results may lead to erroneous conclusions regarding the effective dissolution of U-5 Fs alloy in hot (>700°F) sodium, it is important to point out that the "solubility" of fuel material in the coolant is not to be disregarded. Although no specific measurements of such effects were made during this particular test series, earlier test results in 1965 and 1966 disclosed that exposed U-5 Fs fuel material does indeed tend to scavenge oxygen from the sodium system. Because the results of these tests have not been reported formally, they will be given here.

In the first series of tests the unclad specimen was submerged under hot bulk sodium (at 700°F) for approximately eight months (May 21, 1965 to Jan. 20, 1966). Before its removal from the IBC* the test subassembly was subjected to a thorough cleaning that converted any adhering sodium to NaOH; it was washed with demineralized water, then with 0.01M acetic acid, and again with water. Finally, the subassembly was dried with argon and unloaded from the IBC into the FCF air cell. The specimen and an unirradiated (fresh) fuel pin were transferred to the argon cell for visual and photographic examination.

Visual inspection of the surfaces of the two fuel pins revealed no perceptible difference. No evidence of unusual surface oxidation was indicated. Nevertheless, diameter measurements were made. Table III summarizes the results of these measurements before irradiation and after eight months under the bulk sodium.

TABLE III. Diameter Measurements of a Fuel Pin before Irradiation and after Eight Months of Residence under the Bulk Sodium

	Before Irradiation	After Eight Months of Residence under Sodium
Max diameter, in.	0.1477	0.1450
Min diameter, in.	0.1431	0.1404
Avg diameter, in.[a]	0.1438	0.1421

[a]Overall average of all determinations.

* Interbuilding Coffin: a portable shielded cask for transferring irradiated subassemblies from the reactor building to the Fuel Cycle Facility (FCF).

These measurements revealed a definite taper in the pin, with the maximum dimension associated with the top and the minimum dimension with the lower end.

With suspicion of weight loss clearly indicated, the spade and transition pieces were removed, and the specimen was weighed. Measurements taken with the balance in the junior cave area indicated a decrease of 3.2273 g, i.e., from 65.2327 to 62.0054 g. When and where the loss occurred is not known, but indications point to the mechanical loss of loosely adhering oxide during the washing operation.

The second unclad fuel specimen resided in either the core or storage basket for approximately 13 months (May 21, 1966 to June 21, 1967). In this case a water wash was not used. Dry air was passed through the IBC to oxidize adhering sodium. Following completion of the oxidation, the subassembly was transferred to the air cell for disassembly. After experiencing some difficulty, the unclad specimen was eventually separated from the cluster and placed in a clear plastic bag. (Activity surveys revealed an irradiation field of 14 R/hr beta-gamma on direct contact and 50 mR/hr beta-gamma at 2 ft.)

Following superficial examinations, the specimen was cleaned with methyl alcohol. Closer examination revealed an oxide layer, a portion of which had sloughed into the surrounding bag. Subsequent weighing operations and chemical analysis of the residue in the bag indicated a total weight loss of 4.11 g, i.e., from 69.74 to 65.63 g. The results of these studies indicate an average daily weight loss of 12 mg/day.

Analyses for uranium were conducted for primary sodium samples collected before and after the exposed-fuel material was installed in the core. The results indicated an upper limit of 2.7 mg of uranium per pin per day; yet the actual weight loss from two fuel pins, submerged for a long time under sodium at 700°F, amounted to approximately 12 mg per pin per day. An explanation for this apparent discrepancy involves the mechanism of fuel loss. Probably the pins that were submerged for extended periods "gettered" oxygen from the primary coolant. Oxide layers formed on the fuel; after an unknown degree of accumulation, mechanical sloughing occurred. The resulting particles apparently settled out and were effectively removed from the system.

E. Leakage of Gaseous Activities to the Reactor Building

In the event of a fission-product release from a defective driver or experimental fuel element, it is conceivable that access to the reactor building could be restricted from the leakage of cover gas around seals in the various primary-tank penetrations. In fact, the significant release of

activity to the reactor building has been noted on two previous occasions.[5,6] To correlate the magnitude of an activity release to the reactor building with a known amount of exposed fuel, thereby placing radiological-safety considerations on a firmer quantitative base, a continuous surveillance was made of radiation levels throughout the plant during the tests. Although the results of these tests were negative, primarily because of the small signal generated by the two unclad elements, a documentation of these results is nevertheless worthwhile.

Anticipating similar future tests with more exposed-fuel material, a review of these efforts is worthwhile to document the results of an exhaustive study of the various seals located between the reactor building and the primary tank.[7] Eventually, the results of future exposed-fuel studies, coupled with available information[8] may be used to suggest methods of reducing radiological hazards in the reactor building when fission products are inadvertently released in the core.

In the following description of the various seals, the information compiled by Nobles[8] will be quoted closely and sometimes verbatim.

1. Types of Seals

All penetrations that could permit cover-gas leakage to the reactor building involve two fundamental types of seals. These are: (a) static seals--those having no moving parts; and (b) dynamic seals--those through which a moving shaft passes.

a. Static Seals. All static seals (on the primary-tank cover) are solid, soft copper gaskets, in the form of "O" rings, compressed between steel flanges. Bolt pressure is applied until copper is extruded into minute irregularities of the sealing faces to achieve a gastight seal. Unlike conventional "O" rings, no control of gasket stress is provided by the flanges. This construction permits small adjustments (for alignment purposes) to be made by tightening flange bolts, so that opposing flange faces are slightly out of parallel. Extreme limits of these adjustments are 20 to 30 min of angle, because the probability of leakage increases with flange misalignment. With correct installation and surveillance, this type of seal has been reliable. Some leakage has occurred, but judicious bolt tightening or replacement of the copper ring has been effective. Essentially all static seals on the cover can be considered as not leaking.

b. Dynamic Seals. Through the primary-tank cover and rotating shield plugs there are 29 penetrations for moving shafts; each requires a dynamic seal. Because they accommodate only translational motion, 20 of these dynamic seals are stainless steel bellows; therefore, they can be considered leaktight. The remaining nine dynamic seals must

undergo seal rotation or rotation and translation; thus they are the major source of cover-gas leakage to the reactor building. Tables IV and V list these dynamic seals.

TABLE IV. Dynamic Seals in Small Plug

Technical Designation	Number	Motion[a]	Type of Seal
Control Rod	12	T	SS Bellows
Auxiliary Core Gripper	1	T	SS Bellows
Holddown	1	T	Split Packing
Core Gripper	1	T&R	Split Packing
Cover Lift Column	2	T	Split Packing

[a]Under the column heading "Motion," "T" indicates translation, "R" rotation, and "T&R" a combination of both.

TABLE V. Dynamic Seals in Primary-tank Cover

Technical Designation	Number	Motion[a]	Type of Seal
Reactor-vessel-cover Lock	3	T	SS Bellows
Safety-rod Drive	2	T	SS Bellows
Throttle Valve	2	T	SS Bellows
Storage Basket	1	T&R	Split Packing
Transfer Arm	1	T&R	"O" Ring
Transfer Port	1	R	"O" Ring
Primary Pump	2	R	Labyrinth Seal

[a]Under the column heading "Motion," "T" indicates translation, "R" rotation, and "T&R" a combination of both.

The holddown column, core gripper, and two cover-lift columns on the small plug, and the storage basket column, transfer arm, transfer port, and two primary pump seals in the primary-tank cover allow most of the leakage.

2. Split Packing-gland Seals

Five of the penetrations--the holddown, core gripper, storage basket, and the two cover lifts--have split packing-gland seals. The upper and lower packings of these glands are separated by a floating collar. This collar, which can be pressurized with argon, transmits compressive force and forms an annular void around the shaft. When the shaft is not required to move, as when the reactor is on line, the packings are compressed by a

motorized plate driving a tubular compression ring. Each of these five shafts must slide through its seal during some phase of fuel handling. To permit this, the packing pressure is relaxed, and the seal is said to be "open." When the seal is open, gas cannot be prevented from being carried through the seal by the moving shaft surface. To ensure that this leakage is not from the reactor cover gas, argon is injected into the void formed by the collar between the upper and lower packings. This injected gas leaks both ways, into the reactor cover gas and into the reactor building, but it prevents cover gas from escaping through the seal.

The seal around the transfer arm shaft is generally the same, but the upper and lower seals are "O" rings with argon injected between them.

3. Labyrinth Seals on the Primary Pump Shafts

Glands for the transmission of continuous high-speed rotary motion present a problem, to which the solution is a compromise between necessary running clearance and leakage. Labyrinth seals limit leakage by providing a tortuous path for the leaking gas through the running clearance. Again, as in the split packing-gland seals, gas can be introduced between two labyrinth seals; by leaking both ways, they isolate the inside from the atmosphere. The primary pump motors are hermetically sealed to the primary-tank cover by a welded diaphragm. A split labyrinth seal, with injected argon, isolates the reactor cover gas from the argon inside the motor case. Only with concomitant failure of the diaphragm and the argon-injection system could reactor cover gas escape to the reactor building; nevertheless, these two penetrations are discussed here since they are dynamic seals.

4. Transfer-port Sealing

Subassemblies enter or exit the reactor through a cylindrical plug valve that is opened or closed by a one-quarter turn of the plug about a horizontal axis. The plug is rotated by concentric integral shafts through "O"-ring glands in the end plates of the valve case. Any existing leakage from these two seals has been small enrough to escape detection.

"O" rings in grooves machined in the convex surface of the plug seal the valve ports. Opening and closing the valve slides the "O" rings across the concave surface of the valve body. Because this is a severe usage of the "O" rings, leakage is unavoidable. To minimize the leakage, however, a static seal plug is kept in the top port whenever the transfer port is not in use. Only in the intervals between shield-plug removal and coffin placement (or vice versa) can cover gas leak to the room.

5. Miscellaneous Systems

Fission-gas and reactor cover-gas monitoring is accomplished in a tubing system that involves two of the static sealed nozzles. The system itself, as distinguished from the nozzles, has had detectable leaks into the reactor building.

A wide variety of seals and glands reduce the leakage of argon from the reactor cover-gas plenum to the reactor building. In principle, the high integrity of the various seals is achievable through good maintenance procedures. Nevertheless, because of the operational nature of dynamic gland seals, a finite probability for cover-gas leakage will always exist. Although proper maintenance and surveillance procedures will reduce leakage greatly, they will not completely eliminate it.

6. Surveillance of Radioactivity Levels in the Reactor Building

Suppose that the increased release of gaseous fission products to the cover-gas plenum would be sensed as a general increase in radioactivity levels in the reactor building. Measurements with existing plant-monitoring equipment were made throughout the periods immediately preceding and during the tests. In assessing the importance of leakage effects to the reactor building, three sets of radiation-survey data are needed: (a) activity levels with the reactor shut down; (b) activity levels with the reactor operating normally (i.e., with no exposed fuel); and (c) activity levels with the reactor operating with exposed fuel. Data taken with the reactor operating must be taken long enough after startup to permit the achievement of radioactive equilibrium. Table VI summarizes the results for the various monitoring points. The first column describes the location and nature of the monitoring equipment. Under the second column are the activity levels established at 0445 on Nov. 15. Since the reactor had been shut down for approximately 24 hr and since the cover-gas system had been purged with fresh argon, these values may be used as an approximate measure of unavoidable environmental effects. The third column of Table VI lists average (equilibrium) values established during the period 2400, Sept. 9 to 2400 Sept. 10, when the reactor was operated normally, i.e., with no exposed fuel. The fourth column of Table VI gives values for the activity level registered just before shutdown after running for approximately 53 hr with the exposed fuel.

Inspection of the data in Table VI verifies that little or no significant increase occurred when the reactor was operated with exposed fuel. In only two instances, i.e., for the air monitors located on the main and basement floors, were significant increases noted. However, neither is real: both resulted from inadvertent leakage of cover gas from a faulty fitting in the pumped-gas loop.

TABLE VI. Radiological Survey Data, Reactor Building

Location and Nature of Monitoring Equipment[a]	Environmental Background	Normal Operating Conditions	Exposed Fuel in the Core
East and West NaK Lines	60	135	120
North and South NaK Lines	3.0	9.5	9.0
West Shutdown Cooler	40	80	75
North Shutdown Cooler	20	45	55
Secondary Na Line--Na Boiler	4.0	10	9.5
Na Cold Trap Room	0.2	0.8	1.0
Na Sampling Cell	<0.03	≈0.05	≈0.05
Argon Purification Cell	40	≈0.1	2.5
Argon Receiving Tank	0.2	0.08	0.12
FERD Loop	12	45	45
Transfer Arm	0.5	0.7	0.5
Shield Cooling Filters	<0.05	0.05	0.05
Basement below FERD	10	30	25
Depressed Area Na Lines	0.3	1.0	1.5
Main Floor, Arms, counts/min	100	150	1500
Basement Floor, Arm, counts/min	500	400	2000
Outside Air Monitor, counts/min	20	90	180
Reactor Plant, Personnel Lock	<20	<20	<20
Primary Tank Top	<20	<20	<20
Reactor Plant Secondary Na Line	<20	<20	<20
Shield Cooling Filters	<20	<20	<20
Reactor Plant Operating Floor	<20	<20	<20
Argon Purification Cell	<20	<20	<20
Thimble Cooling Filters	<20	<20	<20
Na Boiler Plant, above Southeast Door	<20	<20	<20
Isolation Trip, above Small Plug Superstructure	<20	<20	<20
Isolation Trip, near Fuel Handling Console	<20	<20	<20
Na Purification Cell, Exhaust Fan	<20	<20	<20
Reactor Plant Exhaust Air	<20	<20	<20

[a]Except where noted otherwise, all units are in mR/hr.

Although the data in Table VI are inconclusive and cannot be used definitively to assess the radiological implications of operating with known fission-product releases, they nevertheless verified that operation with two completely unclad fuel elements produced no perceptible effects.

Care must be taken, however, to avoid the conclusion that the radiological implications of operating with defective fuel elements are trivial. Although the fission-product release rate for metallic fuels is small, that for experimental oxide elements is large. For example, the S/N measured in these tests was approximately 8.

For an actual fission-product release from an oxide element in experimental subassembly XO11, a S/N ratio of 3000 was noted. For a release, such as described by Smith et al.,[5] leakage effects to the reactor building were significant. Obviously, care must be taken to consider differences in the fission-product release rates for metallic and ceramic fuels in studies concerned with the operation of the reactor plant under failed-fuel conditions.

It is apparent also that the signal generated from two unclad metallic fuel pins is insufficient to evaluate the importance of leakage effects. In similar future tests the signal must be increased, either by using a much larger number of unclad fuel pins or by using oxide-fuel specimens.

VII. SUMMARY AND SIGNIFICANCE OF THE EXPOSED-FUEL CALIBRATION STUDIES

Of vital importance in the qualitative and quantitative diagnosis of a given fission-product release is a firmly established experimental base relating the maximum (hypothetical) fission-product release from a single driver-fuel element to the ever-present, unavoidable background. Such a base may be established by measuring the responses of all fission-product monitoring when a known amount of unclad fuel is in the core and by comparing these responses with the normal background levels. Division of the separated exposed-fuel signal by that of the background signal produces a value that is referred to as the S/N (signal-to-noise) ratio. It may be shown mathematically that the S/N ratios for all monitoring systems, in the event of a complete release of all fission products from a driver or an experimental fuel element, should be the same. Furthermore, the S/N ratio defined in this manner should also be independent of time-power history. Thus, in the event of an actual fission-product release, measurements of the S/N ratio provide important information relating to conditions such as the following: whether driver fuel was involved, whether the release was gas or bond type, the size of the defect, or whether exposed fuel was involved.

Accordingly, the responses of all monitoring systems were established when the reactor was operated under full-flow conditions at 50 MWt with and without the presence of two completely unclad fuel pins.

The following single-pin S/N ratios (average values) were found: for the FGM, 4.09; for the FERD, 3.6; for ^{135}Xe in the cover gas as sensed by the RCGM, 1.34; for ^{135}Xe in the cover gas as measured radiometrically, 1.33; and for ^{131}I in the primary coolant, 1.65.

What appears to be a serious discrepancy among the various values, which ideally should be the same, is actually not a discrepancy. At the time of the tests a defective fuel element contained in a control rod (L-462) was preferentially releasing long-lived halogen fission products. More specifically, ^{131}I, ^{133}I, and ^{135}I were being released intermittently to the primary coolant. Although this situation was suspected, it was also established unambiguously that the shorter-lived halogen fission products, which serve as ultimate index species for the FGM and FERD systems, were not being released. Accordingly, the S/N ratios for monitoring systems based on

sensing the longer-lived iodine fission products (or their daughters) would be unduly low, simply because the components originating from the defective fuel element would be erroneously included in the background measurements. In a practical sense the deliberate insertion of unclad fuel material into the reactor core made it possible to conclude that fission products were being released to the coolant from a defective fuel element.

The primary objective of the measurements, i.e., the measurement of the hypothetical maximum S/N ratio of a single driver element, was not compromised by the presence of the defective fuel element. Use of the FGM and FERD values, which were not affected by the defective fuel element, resulted in an average value of 3.9.

Valuable subsidiary information was also derived from the exposed-fuel studies. It was shown that the ratio of power generated in a control rod in its lowest position to that in its fully inserted position was 0.52. Such information was needed to establish the practicability of diagnosing fission-product releases from fuel elements in a control rod.

Environmental background effects were established during the measurements. It was shown that for all monitoring systems such effects are small. Accordingly, there is little incentive to engage in efforts intended for a reduction in environmental background effects.

The dissolution of U-5 Fs fuel material in hot (>700°C) sodium was studied. The results of chemical analyses of coolant samples, indicated that uranium was not present in the coolant to an extent greater than the limits of detection, i.e., approximately 0.2 ppb. Using this value as an upper limit enables corresponding upper limits of 2.7 mg/day per pin to be placed on the dissolution rate of U-5 Fs fuel material.

Finally, concentrated efforts were devoted to attempts to detect significant changes in radiation levels in the reactor building during the tests. The lack of significant change implies that insofar as radiological effects are concerned, the reactor may be operated continuously with at least two (and possibly many more) defective driver-fuel elements in the core.

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