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EG&G Idaho, Inc.
Idaho Falls, Idaho 83401

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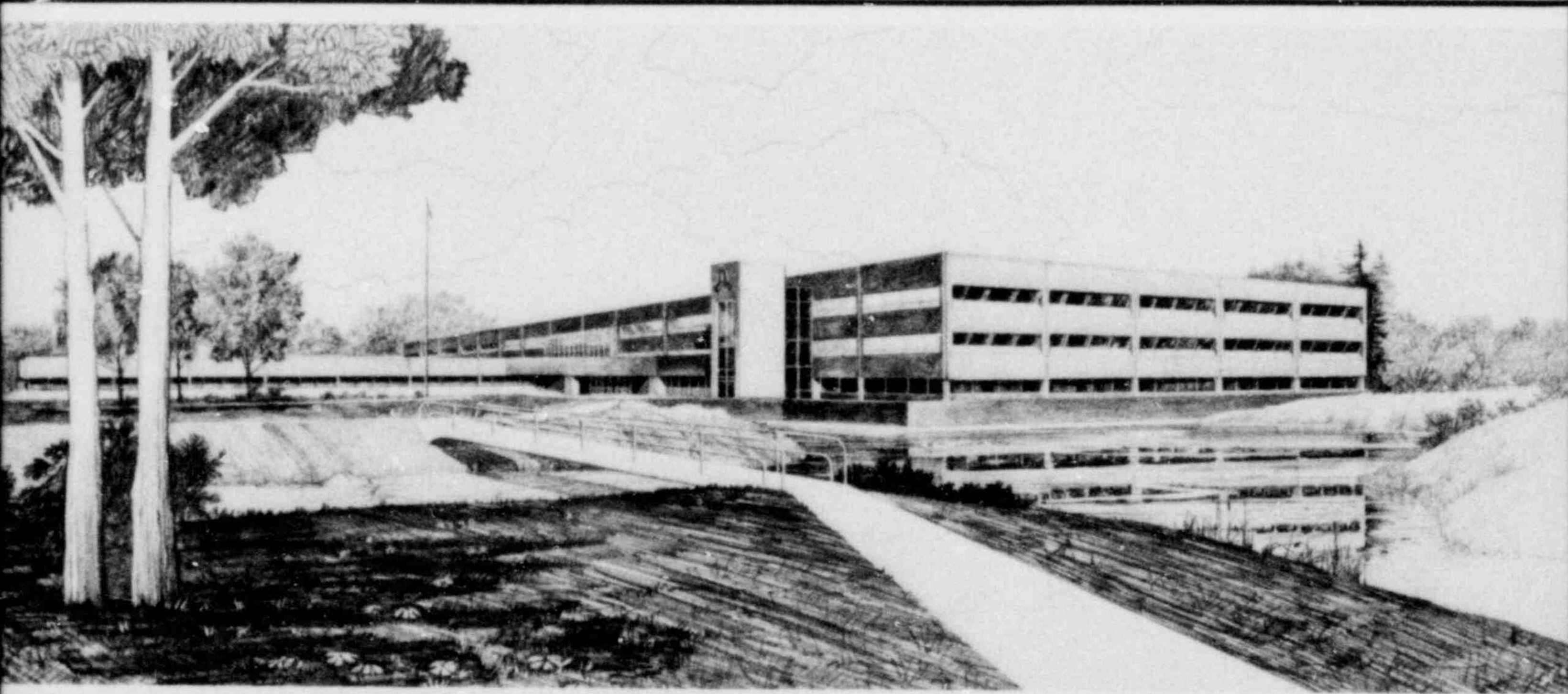
IFA-429 EXPERIMENT UPDATE REPORT

M. K. Charyulu
T. D. Appelhans

U.S. Department of Energy
Idaho Operations Office • Idaho National Engineering Laboratory



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EG&G Idaho

IFA-429 EXPERIMENT UPDATE REPORT

by
M. K. Charyulu
T. D. Appelhans

ABSTRACT

The Halden Test IFA-429, managed by EG&G Idaho, Inc. for the U. S. NRC is designed to study fill gas absorption, fission gas release and thermal behavior of UO_2 fuel. The design, operation and the instrument performance of IFA-429 are described. The conclusions based on the analysis of fuel behavior data at low burnups and the current status of the test are provided. A preliminary analysis of fuel performance data at medium and high burnups is presented. Areas worthy of further analysis are identified and recommendations made.

ACKNOWLEDGEMENT

The authors thank Mr. R. W. Miller, representative of the U. S. Nuclear Regulatory Commission at Halden, for his assistance. The authors also wish to thank the members of the Halden Project for their advice and suggestions and Ms. G. S. Reilly for processing the data.

CONTENTS

SUMMARY

The Instrumented Fuel Assembly 429 (IFA-429), managed by EG&G Idaho, Inc. for the U. S. Nuclear Regulatory Comission, is being irradiated in the Halden Reactor to study helium fill gas absorption, fission gas release, and the thermal behavior of UO_2 fuel. The parameters studied with respect to these areas are power, fuel density, fuel grain size, and pellet-to-cladding gap size.

This report provides an update of the IFA-429 experiment including the instrument performance, a review of helium absorption results, and preliminary analysis of fission gas release data. The fill gas pressure in the fuel rods decreased during early life due to fuel densification and helium absorption by the fuel. Higher rod operating power and large fuel-cladding gap size appear to increase helium absorption, probably due to the high fuel temperature that results. No significant release of fission gases was measured in the fuel rods with internal pressures of 2.58 MPa (STP) which have been operating at rod average linear heat ratings of <25 kW/m for up to 20,000 MWd/t. In one rod, which is suspected to have a lower fill gas pressure (due to leakage through an instrument lead), there appears to have been a significant release of fission gas at $\sim$13,000 MWd/t burnup; the rod had been operating at $\sim$ 38 kW/m with fuel centerline temperatures at $\sim$1250 K prior to the suspected release. The release was initiated during a planned overpower transient during which the rod power was increased by $\sim$40%. The other instrumented fuel rods that have been operating at high power ($\sim$35 kW/m) have shown negligible fission gas release at burnups to 34,000 MWd/t.

The following recommendations are made: (a) The assembly continue to operate to burnups of $>$50,000 MWd/t; (b) the upcoming postirradiation examination (PIE) of the rod which indicated sudden increase in temperatures focus on fission gas release; (c) the ultrasonic thermometer fuel temperature data (hitherto unused because of calibration difficulties) be evaluated for use as a relative fuel behavior indicator; and (d) detailed analysis of the fission gas release be performed using the GRASS-SST computer code.

I. INTRODUCTION

The fuel behavior research program of the United States Nuclear Regulatory Commission (USNRC) includes in-pile and out-of-pile experiments which are conducted to study fuel rod responses to normal and abnormal operating conditions. The data obtained from these experiments are used in the development and verification of reactor safety analysis codes. Included in this program are studies of uranium oxide and mixed oxide fuel behavior performed at the Halden Boiling Water Reactor (HBWR) in Halden, Norway. The HBWR was built by the Norwegian Institute for Atomenergi, and is operated by the organization for Economic Cooperation and Development (OECD) Halden Reactor Project through an international agreement between the participating governments and organizations.

Experiments performed at Halden include short and long-term irradiations of instrumented fuel assemblies, and produce a variety of fuel rod performance and reliability data. One of these experiments, managed by EG&G Idaho, Inc. for the USNRC and termed the Instrumented Fuel Assembly 429 (IFA-429), was designed to study fill gas absorption, fission gas release, and fuel thermal behavior of UO_2 fuel. The parameters studied with respect to these areas are power, burnup, fuel density, fuel grain size, and pellet-to-cladding gap size.

The objective of this report is to document the current status of IFA-429 and provide a preliminary analysis of selected data. The helium absorption results[1] and the IFA-429 data[2] from June 1975 to June 1978 have been previously reported. This report highlights the significant areas of the IFA-429 data acquired from June 1975 (beginning-of-life) to December 1979 (up to 30,000 Mwd/t burnup).

Section II of the report describes the design and operation of IFA-429 and the instrument performance. Section III contains selected fuel temperature and fuel rod pressure data, and a discussion on apparent significant release of fission gas in one of the IFA-429 fuel rods. Section IV contains the concluding remarks and recommendations.

II. DESIGN AND OPERATION OF IFA-429

IFA-429 was designed to study the absorption of helium fill gas by uranium oxide fuel, steady state and transient operating effects on fission gas release, and fuel thermal behavior after extensive fuel burnup. Data concerning these phenomena are being obtained with respect to fuel density, fuel grain size, as-fabricated fuel-cladding gap size, fuel rod power, and fuel burnup. Gas release and helium absorption are monitored in-pile by measuring rod internal pressure of selected fuel rods, while the thermal behavior of the fuel is monitored with fuel centerline temperature instruments. Information on fuel crack patterns, fuel structure, and internal gas distribution are obtained from postirradiation examinations of fuel rods, which have been removed (and replaced with new rods) from the assembly at various burnups. The test design and the instrument performance are described in the following sections.

1. TEST DESIGN

IFA-429 consists of 18 fuel rods arranged in three 6-rod hexagonal clusters (46-mm pitch, aligned in a vertical stack configuration) as shown in Figure 1. Each cluster is 586 mm long, with each fuel rod containing a 244 mm long fuel column, as shown in Figure 2.

In order to provide the capability to perform fast power ramps with the assembly, each fuel rod cluster is surrounded by a neutron absorber shield. The shields, made of silver encased in stainless steel, as shown in Figure 3, are mechanically linked to each other and can be hydraulically raised and lowered, in approximately 5 and 14 seconds respectively. In the down (normal) position the sleeves surround the fuel rod clusters and decrease the average incident thermal neutron flux by approximately 30%. The distribution of the fuel rod cluster positions in relation to core axial position, and their steady state power profiles are shown in Figure 4. The average

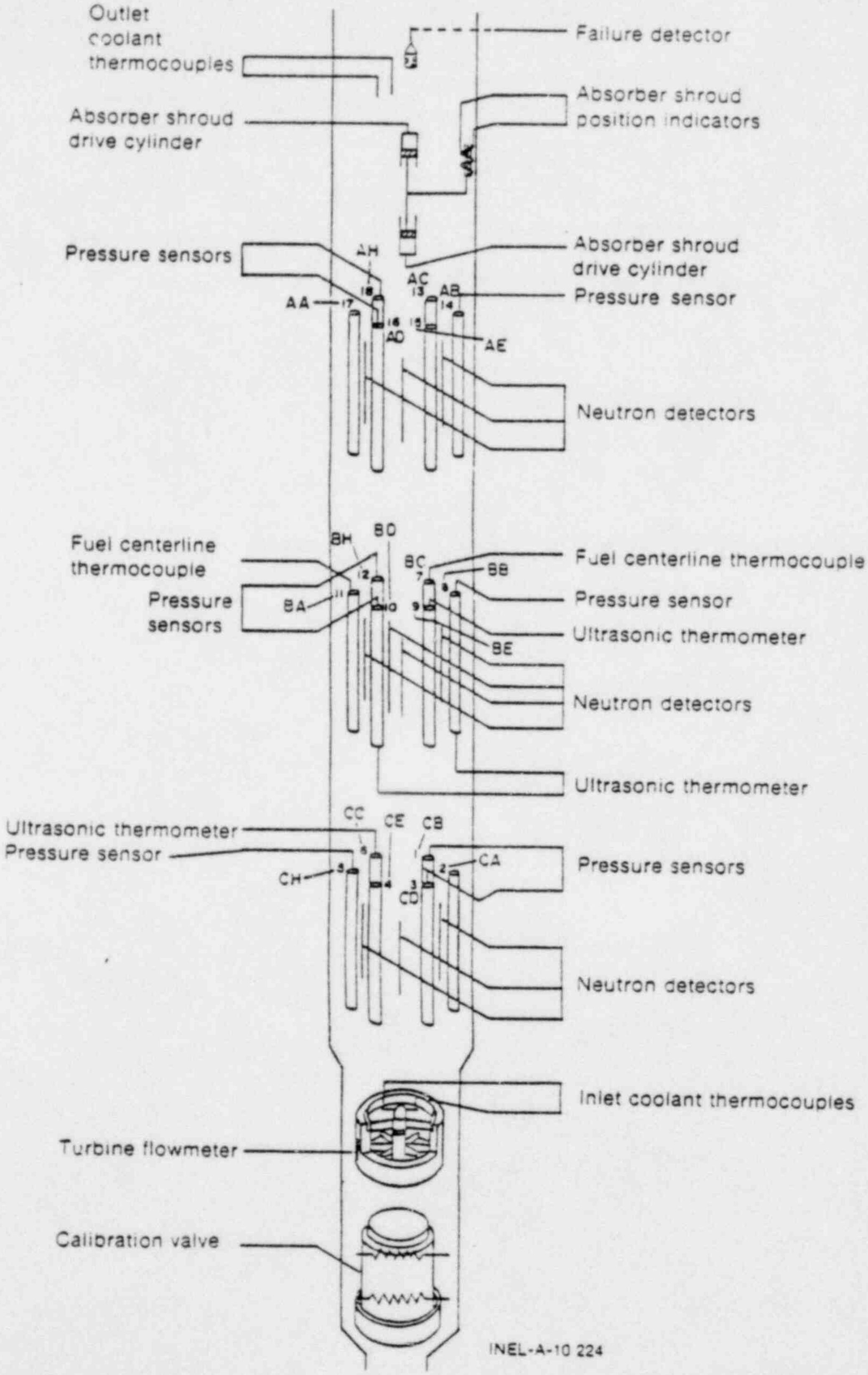

Figure 1. IFA-429 experiment assembly.

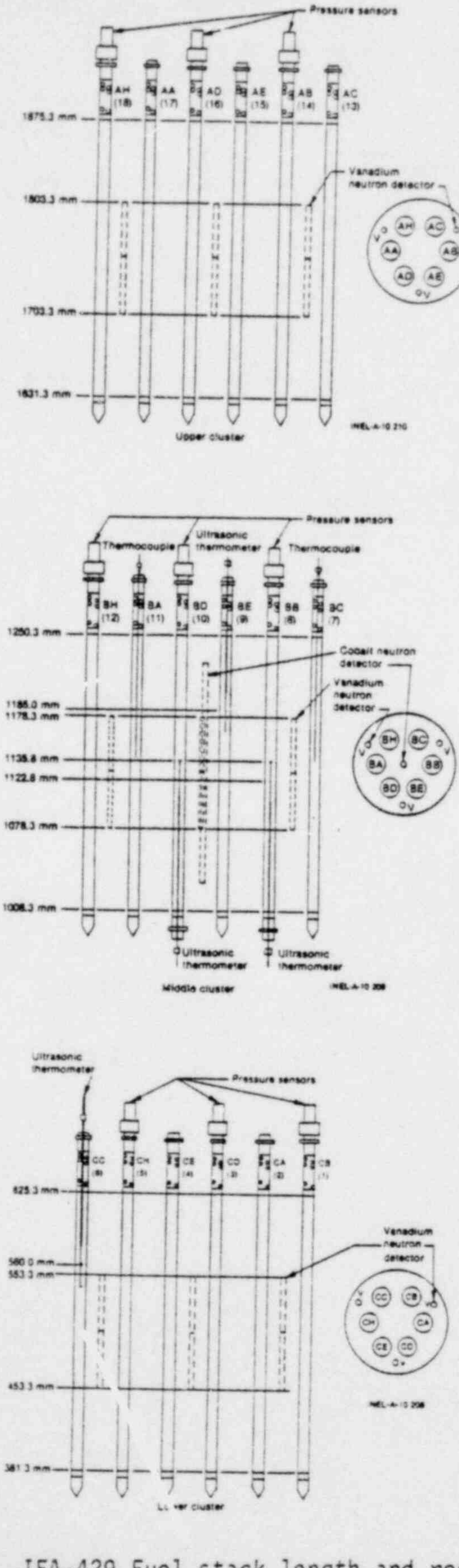

Figure 2. IFA-429 Fuel stack length and rod instrumentation for the three clusters with respect to the HBWR lower core plate.

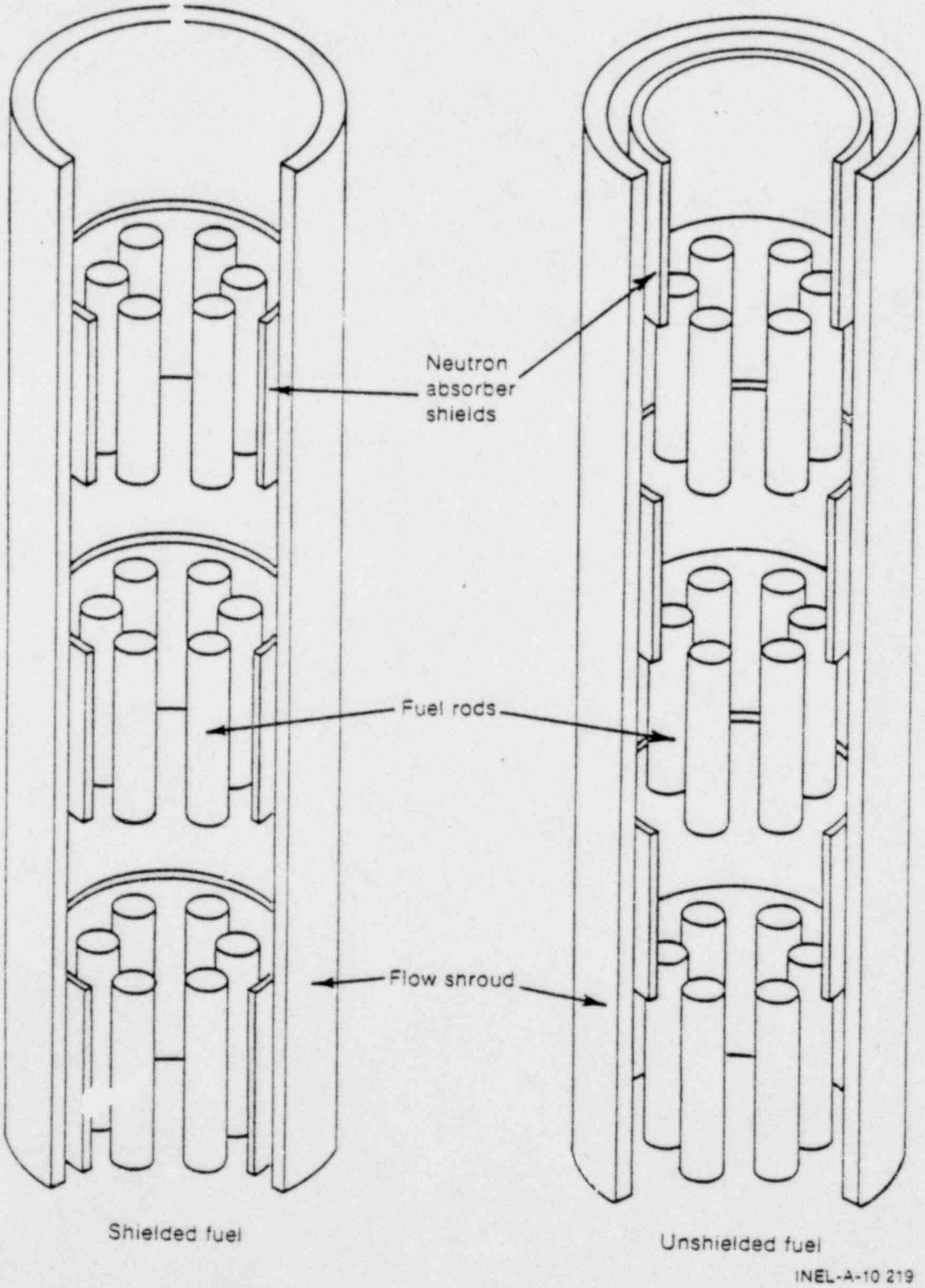

Figure 3. IFA-429 assembly showing neutron absorber shields.

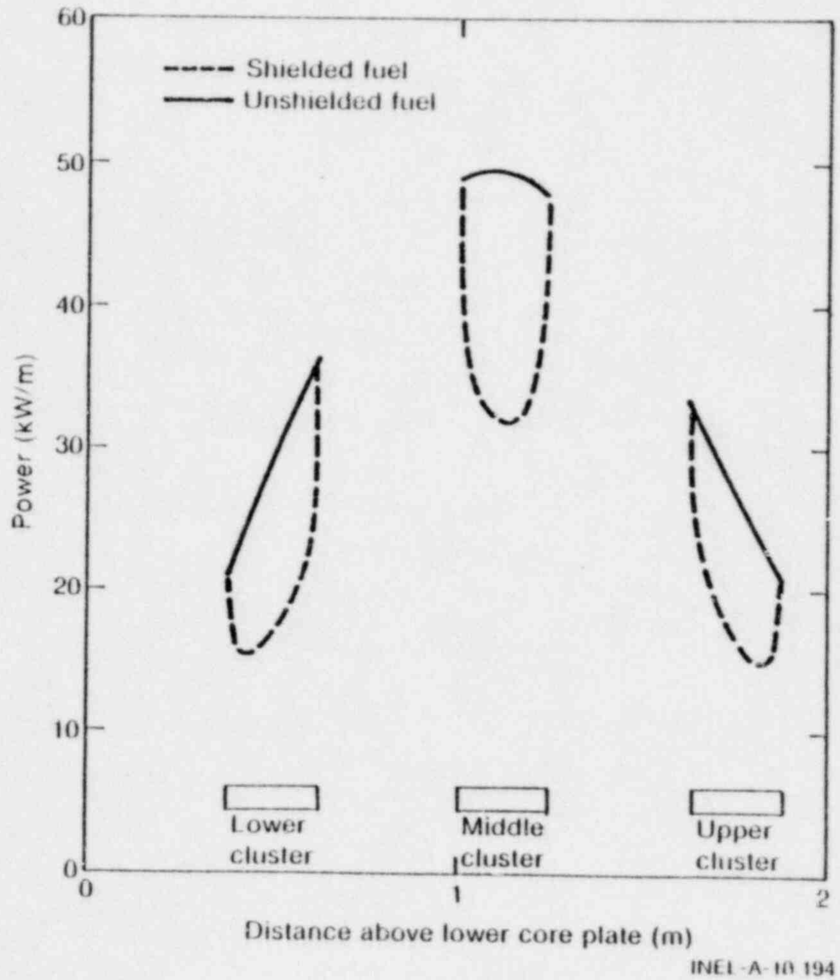

Figure 4. IFA-429 power profile during normal (shielded) and high power (unshielded) operation.

power in the lower cluster, the middle cluster, and in the upper cluster is approximately 19 kW/m, 35 kW/m and 18 kW/m, respectively, at full reactor power with the shields in the down (shielded) position. In the up (unshielded) position the powers are about 40% higher.

The design parameters of the 18 fuel rods used in the initial loading of IFA-429, and the 15 replacement rods are listed in Tables 1 and 2, respectively. The pressed and sintered uranium oxide fuel is enriched to 13 wt% ^{235}U. The fuel pellets were fabricated into one of three densities (91, 93, or 95% theoretical), and one of two mean grain sizes (approximately 17 or 5.9 μm). The fuel pellets are dished on each end, clad in zircaloy-4 tubes (10.7-mm outside diameter, 0.6-mm wall thickness), and prepressurized to 2.58 MPa with helium. Characterization of axial variations in fuel pellet density and diameter, cladding inside and outside diameter, cladding profile, and pellet-cladding gap size are presented in Appendix A of Reference 2.

The upper and middle clusters contain fuel rods of identical design (with a fuel-cladding gap size of 0.2 mm and fuel grain size of 17 μm) but different fuel densities, with both clusters containing two rods each of 91, 93, and 95% theoretical density (TD) fuel. The lower cluster contains six rods of 95% TD fuel, two rods identical to the 95% TD rods in the upper cluster, two with the larger fuel-cladding gap size (0.36 mm), and two with the smaller grain size (5.9 μm) fuel, as given in Table 1. Each variation in the parameter distribution involves two fuel rods. For example, the lower cluster contains two rods with large gaps, two rods with small fuel grain size, and two rods of nominal design with 95% TD fuel. Such distribution results in each cluster consisting of three pairs of replicate fuel rods.

This distribution of design parameters yields data concerning:

(1) The effect of fuel density variation for different steady state power levels (upper, middle, and lower clusters)

TABLE 1. IFA-429 INITIAL LOAD FUEL ROD DESIGN VARIABLES

Rod	Rod Number	Cluster	Fuel Density (% TD)	Steady State Power (kW/m)	Diamentral Gap Size (mm)	Grain Size (μm)	Instrumentation PS[a]	UT[b]	TC[c]
AA	17	Upper	91		0.20	17			
AB	14	Upper	91		0.20	17	X		
AC	13	Upper	93	18	0.20	17			
AD	16	Upper	93		0.20	17	X		
AE	15	Upper	95		0.20	17			
AH	18	Upper	95		0.20	17	X		
BA	11	Middle	91		0.20	17			X
BB	8	Middle	91		0.20	17	X	X	
BC	7	Middle	93	35	0.20	17			X
BD	10	Middle	93		0.20	17	X	X	
BE	9	Middle	95		0.20	17		X	
BH	12	Middle	95		0.20	17	X		
CA	2	Lower	95		0.20	17			
CB	1	Lower	95		0.20	5.9	X		
CC	6	Lower	95	19	0.36	17		X	
CD	3	Lower	95		0.20	17	X		
CE	4	Lower	95		0.20	5.9			
CH	5	Lower	95		0.36	17	X		

Fuel: Form -- pressed and sintered UO_2 pellets
Enrichment -- 13 wt% U-235
Shape -- length- 15.2 mm
diameters- 9.296 and 9.114 mm
ends- dished to 0.33 mm with a radius of curvature of 16.8 mm

Rods: Fuel stack length -- 244 mm
Plenum length -- 35.4 mm
Fill gas -- helium
Pressure -- 2.58 MPa
Diametral gaps -- 0.20 and 0.36 mm

Cladding: material--zircaloy-4
outside diameter--10.72 mm
inside diameter--9.5 mm.

a PS -- Diaphragm pressure sensor.
b UT -- Ultrasonic thermometer.
c TC -- W5%Re/W26%Re thermocouple.

TABLE 2. IFA-429 EXCHANGE AND SPARE FUEL ROD DESIGN VARIABLES

Rod	Fuel Density (% TD)	Diametral Gap Size (mm)	Nominal Grain Size (μm)
DA	91	0.20	17
DB	93	0.20	17
DC	95	0.20	17
DD	91	0.20	17
DE	95	0.36	17
DH	95	0.20	5.9
DK	95	0.36	17
DM	95	0.36	5.9
DS	93	0.20	17
EA	95	0.20	5.9
EB	95	0.36	5.9
EC	91	0.20	17
ED	91	0.36	17
EE	95	0.20	17
EH	95	0.36	17

Fuel: Form -- pressed and sintered UO_2 pellets
Enrichment -- 13 wt% U-235
Shape -- length- 15.2 mm
diameters- 9.296 and 9.144 mm
Ends- dished to 0.33 mm with a radius of curvature of 16.8 mm.

Rod: Fuel stack length -- 244 mm
Plenum length -- 25.4 mm
Fill gas -- helium
Pressure -- 2.58 MPa
Diametral gaps -- 0.20 and 0.36 mm

Cladding: Material--Zircaloy-4
Outside diameter--10.72 mm.
Inside diameter--9.5 mm.

(2) The effect of grain size variation at constant power (lower cluster)

(3) The effect of initial pellet-cladding gap size at constant power (lower cluster)

(4) The effect of power increases on fuel behavior for different fuel densities (upper, middle, and lower clusters).

The test assembly is designed so that the rods may be removed and replaced with fresh fuel rods. Upon removal, the rods undergo postirradiation examination to determine fuel helium absorption, fission gas content and release, pellet crack patterns, and other structural and material changes. Fuel rod replacements to date have been: (a) February 1976; Rods AA, AE, and CA were replaced with Rods DA, DC, and DK, respectively, after approximately 3600 MWd/t burnup, (b) August 1977; Rods AC and CE were replaced with Rods DB and DH, respectively, after approximately 10,300 MWd/t, and (c) January 1980; Rods BA and BE were replaced with Rods DD and EE, respectively, after approximately 30,000 MWd/t burnup. The fuel loading as of March 1, 1980 is given in Table 3. A series of postirradiation examinations have been performed on Rods AA, AE, and CA, and the results previously presented.[3] Rods AC, CE, BA and BE will undergo post irradiation examination in 1980-81.

IFA-429 began irradiation on June 8, 1975. Since then overpower transients have been performed, by raising the neutron absorber shields, 9 times: September 5, 1975, August 10, 1976, January 21, 1977, April 27, 1977, August 22, 1977, January 5, 1978, August 27, 1978, January 3, 1979 and April 27, 1979. The duration of these tests were varied between approximately one to three hours. The fuel rod pressures and temperatures are recorded before, during and after these transients to assess the effects of the transients on the fuel behavior.

TABLE 3. IFA-429 FUEL LOADING AS OF MARCH 1, 1980

Rod	Rod Number	Cluster	Fuel Density (% TD)	Steady State Power (kW/m)	Diamentral Gap Size (mm)	Grain Size (μm)	Instrumentation PS[a]	UT[b]	TC[c]	Burnup GWd/t
DA	17	Upper	91		0.20	17				~17
AB	14	Upper	91		0.20	17	X			~20
DB	13	Upper	93	18	0.20	17				~10
AD	16	Upper	93		0.20	17	X			~20
DC	15	Upper	95		0.20	17				~16
AH	18	Upper	95		0.20	17	X			~21
DD	11	Middle	91		0.20	17				~0
BB	8	Middle	91		0.20	17	X	X		~34
BC	7	Middle	93	35	0.20	17			X	~34
BD	10	Middle	93		0.20	17	X	X		~34
EE	9	Middle	95		0.20	17				~0
BH	12	Middle	95		0.20	17	X			~33
DK	2	Lower	95		0.36	17				~14
CB	1	Lower	95		0.20	5.9	X			~18
CC	6	Lower	95	19	0.36	17		X		~19
CD	3	Lower	95		0.20	17	X			~18
DH	4	Lower	95		0.20	5.9				~9
CH	5	Lower	95		0.36	17	X			~19

Fuel: Form -- pressed and sintered UO_2 pellets
Enrichment -- 13 wt% U-235
Shape -- length- 15.2 mm
diameters- 9.296 and 9.114 mm
ends- dished to 0.33 mm with a radius of curvature of 16.8 mm

Rods: Fuel stack length -- 244 mm
Plenum length -- 35.4 mm
Fill gas -- helium
Pressure -- 2.58 MPa
Diametral gaps -- 0.20 and 0.36 mm

Cladding: material--zircaloy-4
outside diameter--10.72 mm
inside diameter--9.5 mm.

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a PS -- Diaphragm pressure sensor.
b UT -- Ultrasonic thermometer.
c TC -- W5%Re/W26%Re thermocouple.

2. INSTRUMENT PERFORMANCE

The fuel rod instrumentation used in IFA-429 is designed to measure fuel centerline temperature and fuel rod internal gas pressure. A brief description of the fuel rod assembly instrumentation and the instrument performance is provided in this section. A detailed description of the instrumentation may be found in Reference 2.

2.1 Temperature Transducers

There are two types of fuel centerline temperature transducers used in IFA-429, thermocouples and ultrasonic thermometers.

Two thoria insulated, W5%Re/W26%Re thermocouples (TC's) are inserted in Rods BA and BC from the top of their fuel stacks, as shown in Figure 2. The 1.59-mm-diameter TC's fit into the holes drilled through the center of the fuel pellets (approximately 2-mm dia.), and are located 115 mm from the top of the fuel stack. The centerline TC in Rod BA became erratic[4] on August 22nd 1977, and was disconnected.[5] The thermocouple in rod BC continues to function.

Rods BD, BE, and BB in the center cluster and Rod CC in the lower cluster contain fuel centerline ultrasonic thermometers[3] (UT's). A decalibration problem which had been identified earlier, has prevented use of the UT's. All four of the UT's were disconnected on July 4, 1978, but were reconnected,[6] on request from EG&G, on September 14, 1978.

2.2 Pressure Transducers

Nine fuel rod internal pressure transducers, three in each cluster, are used in IFA-429 (Figure 2). The instruments are contained in, and affixed to, the fuel rods through the upper end plugs. The pressure sensor consists of a platinum diaphragm exposed to fuel rod pressure on one side and to a chamber, in which the pressure can be controlled from outside the reactor, on the other side. The diaphragm rests on an electrical contact on the controllable pressure side. The pressure in the rod is determined by increasing the controlled pressure until it equals that of the rod internal pressure. Inconsistencies in the pressure data led to a need for reproducibility tests (October 5, 1977) during which a fault was discovered. No information is available to indicate how long the system had been faulty. After the fault was corrected another check, which confirmed that the problem had been solved, was performed[5] on November 4, 1977. The results indicated that for the seven transducers, on fuel rods with pressures in excess of 4 MPa (at power), the standard deviation was less than 1%. The two rods with low pressures showed higher deviations, viz, 2% at about 1 MPa and 25% at under 0.3 MPa. The absolute pressures are accurate to within $\pm$0.15 MPa.

2.3 Effects of Thermocouple and UT Lead Design on Fuel Rod Pressure

Following the installation of the assembly in the reactor, it became evident that the pressures in Rods BB and BD were much lower than when fabricated. It appeared that the fill gas had leaked out through the UT leads. This assumption was based on the fact that all welds had been leak checked during fabrication and found to be sound, and that none of the seven rods that contained only pressure transducers (no TC's or UT's) had leaked. Thus it is suspected that all rods with TC or UT instruments leaked, prior to nuclear operation. This will be checked in the FY-80 PIE in which rod BA (a rod with a TC) will be examined.

III. A CURSORY REVIEW OF FUEL TEMPERATURE AND FUEL ROD PRESSURE RESULTS

This section provides a preliminary analysis of the fuel behavior with respect to gap pressure in terms of helium absorption during early life, fission gas release and fuel thermal behavior. The fuel rod plenum pressure analysis is based on the 7 fuel rods that remained at design pressure and the temperature analysis is based on the middle cluster rods which contain thermocouples.

1. HELIUM ABSORPTION

The objective of early life measurement of rod internal pressure was to assess the extent of helium absorption. Vinjamuri and Owen[1] have reported these data previously and only a summary is provided below, for completeness. Figures 5 through 7 show pressure response of five of the seven rods. It can be seen that the fuel rod pressures generally decreased during the first several months, during which time the fuel rods operated at steady-state high powers (although during this time there were a number of reactor power cycles and shutdowns of short duration). The rate of pressure decrease slowed down after the pressures reached a minimum at 3000 to 6000 MWd/t burn-up after which the pressures began to increase, gradually. The early life pressure decrease apparently resulted from both fill gas absorption and fuel densification. The former reduced the quantity of fill gas and the latter increased the rod internal free volume. The rate of pressure decrease leveled off as both the helium absorption sites and the fuel densification effects saturated.

In the upper and lower cluster fuel rods the fission gas release had an insignificant effect on fuel rod pressure during early life. However, helium absorption accounted for 34 to 100% of the pressure decreases. The effect of fuel density on early life pressure is illustrated in Figure 6 where pressures in Rods AH (95% dense) and AB (91% dense) are compared. The greater pressure drop in the

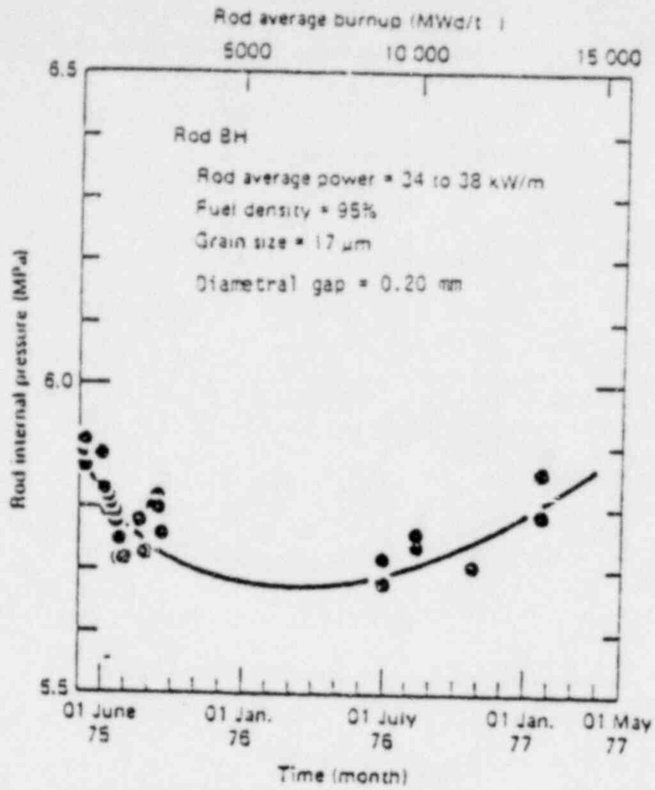

Figure 5. Internal pressure behavior of fuel Rod BH.

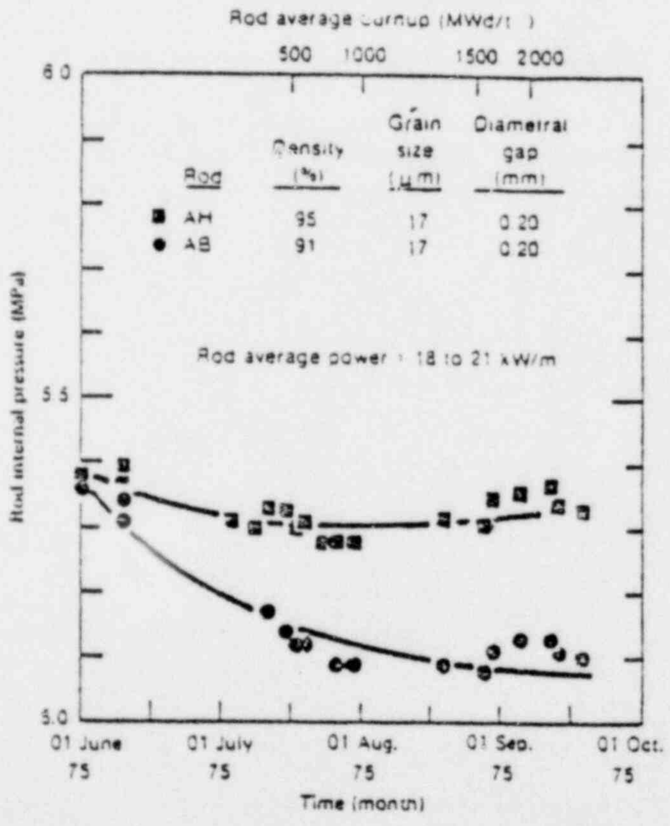

Figure 6. Effect of fuel density on fuel rod internal pressure.

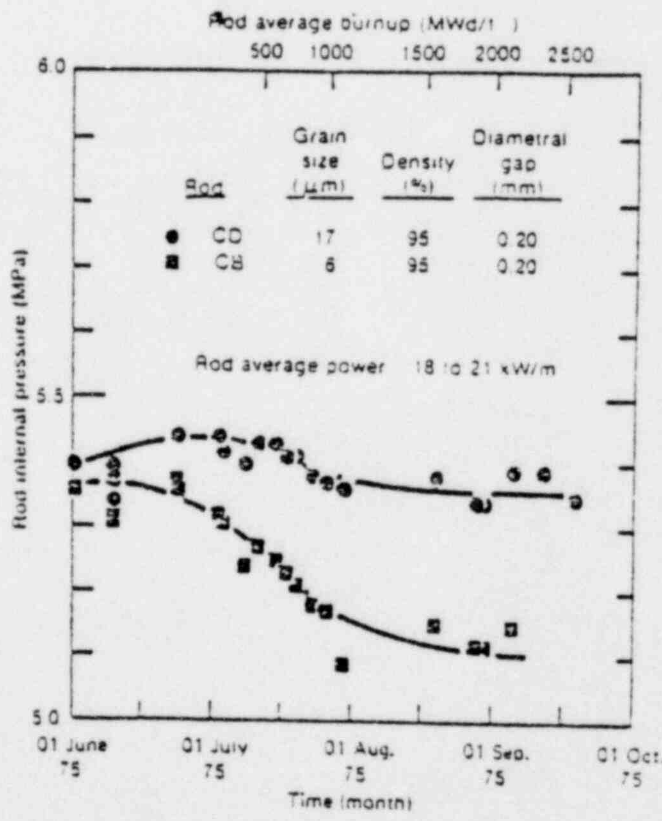

Figure 7. Effect of UO_2 grain size on fuel rod internal pressure.

low-density fuel rod is probably due to enhanced densification. Post irradiation helium content measurements on the identical companion Rods AE and AA indicated similar helium contents. Increased helium absorption in smaller grained fuel is observed in Figure 7. This is attributed to larger grain boundary area in fine-grained fuel. Most of the enhanced pressure decrease in the 5.9 μm fuel grain (Rod CD) relative to the 17 μm fuel (Rod CB) is attributed to helium absorption rather than fuel densification, since at the 95% density level and with the stable microstructure, densification is not a strong function of grain size.

The pressure data indicate that the helium absorption in large-gap fuel rods is greater than in small-gap rods. This greater absorption may be due to the higher fuel temperatures in the large-gap rod. However the data are not conclusive and additional analyses are needed.

2. FISSION GAS RELEASE IN HIGH POWER RODS

The pressure measurements from both the upper and the lower cluster rods, the steady state power levels for which are between 18 and 21 kW/m, have shown (Table 4) that the fission gas releases in these rods have been negligible.

Among the six middle cluster rods the only rod which maintained full pressure is rod BH. Table 5 compares the average pressure for the three high power rods BB, BD and BH for the periods covering November 1977 to January 1978 and April 1979 to August 1979. These data show that, between the two periods, there have been small pressure increases in Rods BB and BH, and no pressure increase in Rod BD.

The internal pressure of rods BA and BC (centerline TC, no pressure sensor) are expected to be of the same magnitude as those of BB and BD, as all rods with TC's are suspected to have leaked early in life. The temperature response of the rods BA and BC (containing TC's

TABLE 4. PRESSURE CHANGE IN IFA-429 LOW POWER RODS

Period	Rod Name	Burnup GWd/t	Power Level kW/m	Pressure MPa	Standard Deviation in Pressure (MPa)
11/77	AB	13	15	4.94	*
to	AD	13	15	5.02	*
1/78	AH	13	15	5.17	*
4/79	AB	19	15-16	4.89	0.07
to	AD	19	15-16	4.96	0.010
8/79	AH	19	15-16	5.17	0.05
1/77	CB	11	17	5.08	*
to	CD	11	17	5.40	*
/78	CH	12	17	5.56	*
4/79	CB	17	17-18	4.98	0.06
to	CD	17	17-18	5.32	0.08
8/79	CH	17	17-18	5.38	0.13

* Estimated to be less than 0.1.

TABLE 5. PRESSURE CHANGE IN IFA-429 HIGH POWER RODS

Period	Rod Name	Burnup GWd/t	Power Level kW/m	Pressure MPa	Standard Deviation in Pressure
11/77	BB	21	25-29	0.36	0.03
to	BD	22	24-28	1.40	0.06
1/78	BH	21	26-31	6.10	0.06
4/79	BB	31	24-28	0.48	0.08
to	BD	32	24-28	1.41	0.09
8/79	BH	30	24-30	6.31	0.16

only) are expected to be very similar to each other since they are suspected to have leaked at about the same rate (as the rods containing UT's). However, as shown in Figure 8, after a burnup of abou 13,000 MWd/t the centerline temperature of rod BA differed significantly from that of BC. Figures 9 and 10 show the power and centerline temperature history of the rods BA and BC during September 1975, and during May 1977, respectively; Rod BC stays at the same temperature, between the two periods, while the temperature in Rod BA increases by nearly 600 K. The temperature trace of Rod BA shows that the TC in Rod BA worked well through this period. The possibility of a significant release of fission gas in Rod BA is, therefore, hypothesized. The hypothesized release appears to have been initi ted during the power transient at 13,500 MWd/t burnup. Because Rod BA does not have a pressure transducer, this hypothesis will be confirmed by PIE in 1980-81. Also, the pressure data available from Rod BH cannot be applied directly to Rod BA as Rod BH has remained at design pressure.

In order to assess the possibility of fission gas release in Rod BA, several preliminary fission gas release calculations were performed using current steady-state and transient fission gas release models. A brief excerpt from Kolstad's paper[7], which reviews 11 steady state fission gas release models currently under use, is given in Appendix A. It provides a brief comparative evaluation of the models to show the present state-of-the-art in correlating fission gas release to local temperatures and burnup (or time).

In Figure 11, the temperatures measured in Rod BA are compared with the Vitanza model[8] that defines the boundary for initiation of fission gas release as a function of temperature and burnup. This curve represents an empirical steady state relationship developed by Vitanza. It can be seen from this figure that the fuel thermocouple temperatures in Rod BA exceeded the fission gas release threshold temperature on January 21, 1977 indicating that fission gas release could be expected. The Vitanza model (see Figure 12) predicts

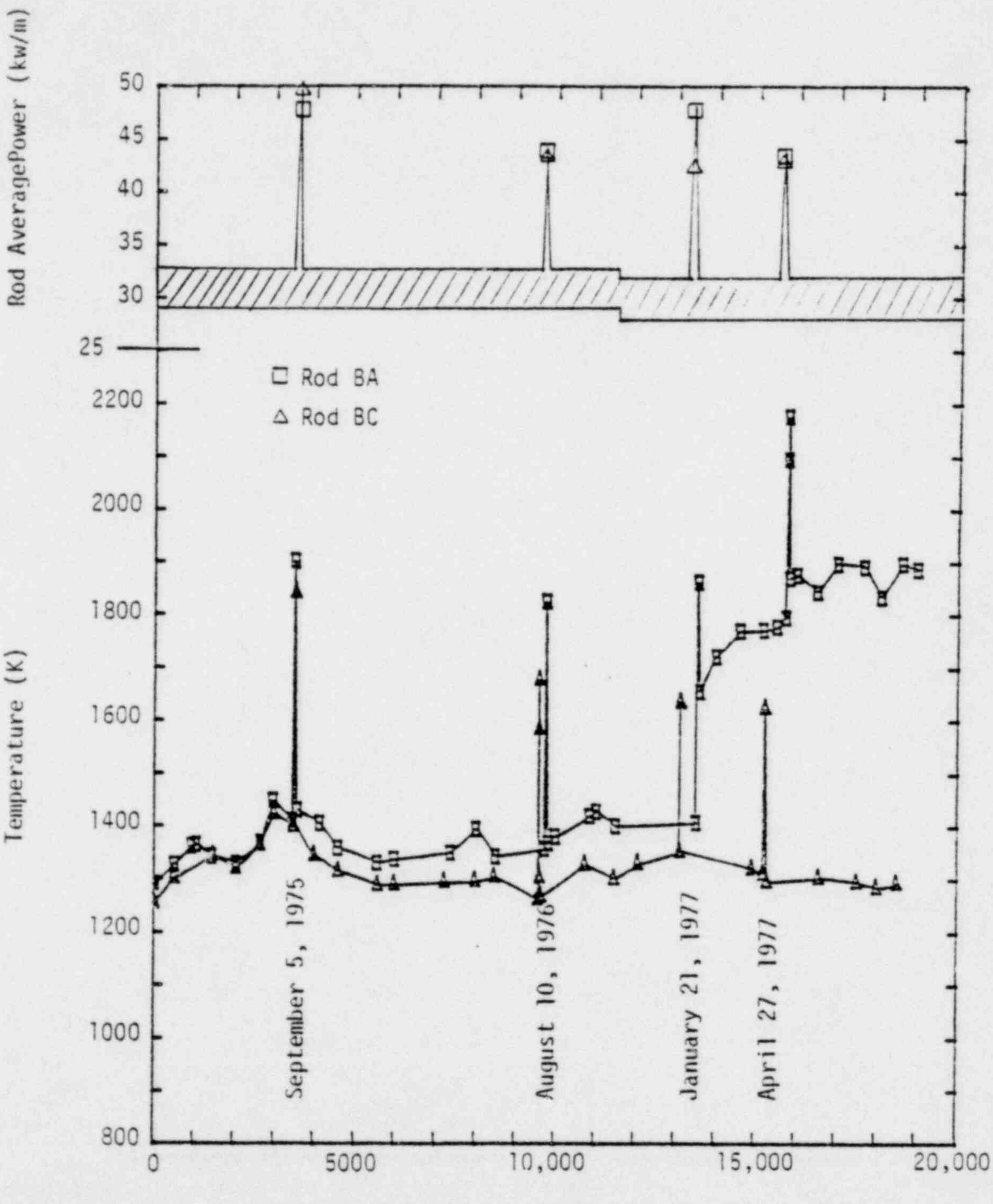

Figure 8. Fuel centerline temperature versus rod average burnup, Rods BA and BC, at power between ~28 and 33 kw/m.

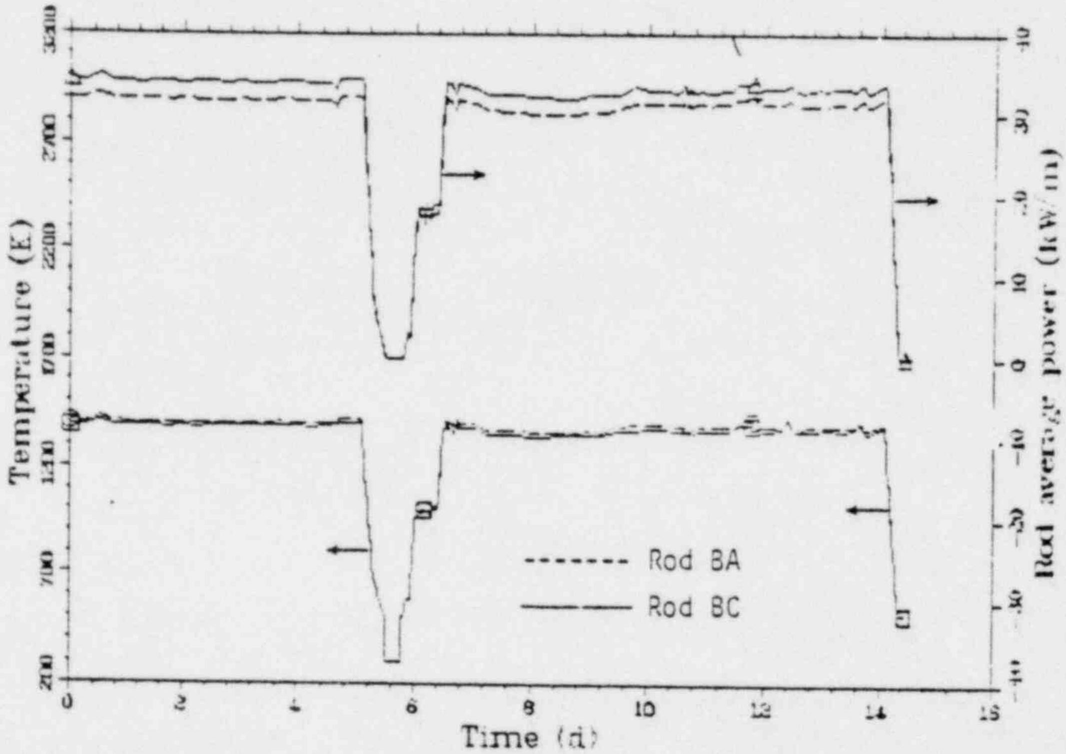

Figure 9. Power and fuel centerline temperature history of Rods BA and BC during September 1975.

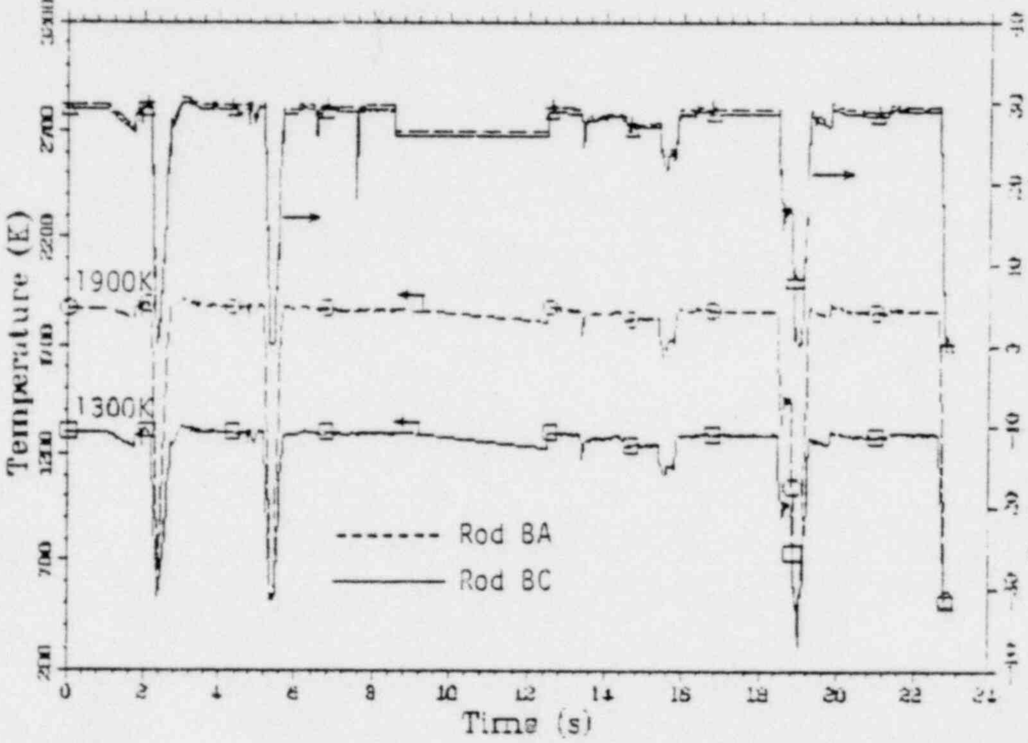

Figure 10. Power and fuel centerline temperature history of Rods BA and BC during May 1977.

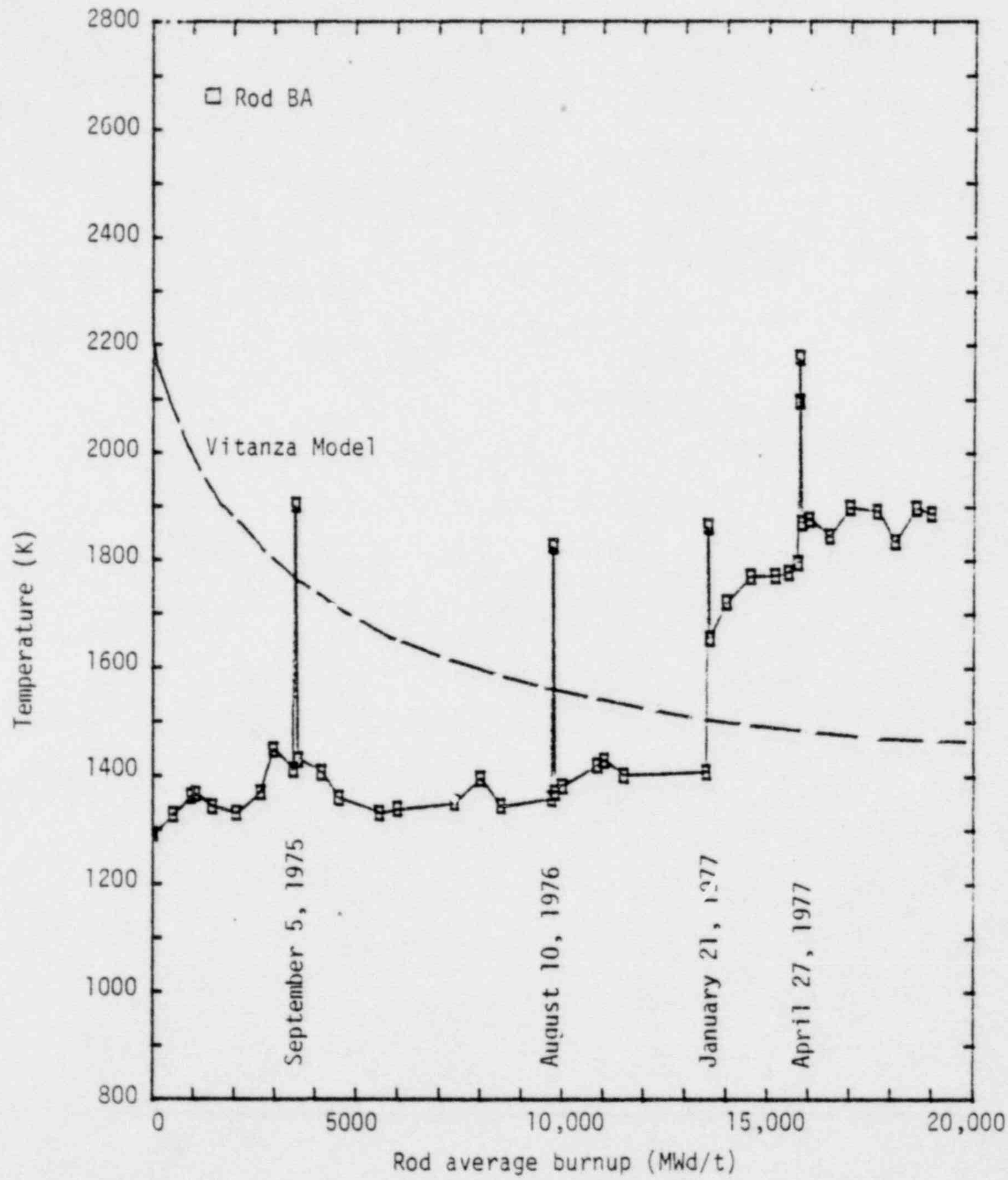

Figure 11. Fuel centerline temperature history of Rod BA and Vitanza[8] threshold for fission gas release as a function of burnup.

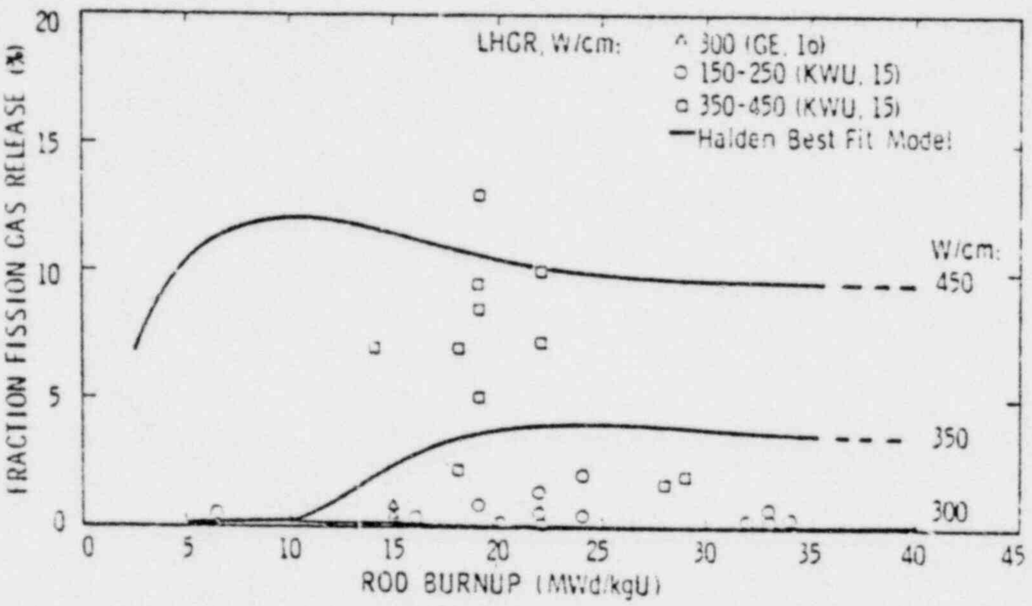

Figure 12. BWR power reactor fission gas release data with Halden best fit model (axial power factor equal to 1.1).

approximately 7% gas release at a steady state power level of 40 kW/m and burnup of 13,000 MWd/t. Based on a fuel centerline temperature of ~1250 K, the other steady state models (reviewed in Appendix A) predict a gas release up to ~5%, except FRAP-S, which predicts a release of ~15%.

In addition to the steady state models, two models which include some transient effects were used to calculate the expected fission gas release. The two models used were Vilpponen's[9] and the GRASS-SST.[10] Vilpponen's estimates (Table 6), based on ramp tested Halden fuel rods with a burnup of 10,000 MWd/t and ramp powers held for less than 6 hours, show that for a base power of 40 kW/m and a ramp power of 60 kW/m the expected fission gas release is ~25%. GRASS-SST, the Argonne National Laboratory (ANL) transient fission gas release code which considers the dependence of fission gas release on the rate of change of temperature across the pellet during a ramp, predicts a fission gas release of ~1%. However, it must be noted that due to the rigidity of its present driver code, several simplifying assumptions had to be made in the input for GRASS-SST, and the results must be used cautiously.

Thus, the fission gas release in Rod BA is predicted to be between 1 and 25% by the various models.

The fission gas release can also be estimated using the measured fuel temperatures in Rod BA. As shown in Figure 8, the fuel temperature in Rod BA increased by ~300 K from just before to just after the power ramp of January 21, 1977, and had increased by ~500 K following the power ramp of April 27, 1977. Figure 13 shows the fuel temperature dependence of the amount of Xe in the fill gas based on IFA-431,[11] and GAP-CON 2.1 test data.[12] From Figure 13 it is estimated that the fill gas would have to contain greater than 80% Xe to drive the fuel temperatures up in Rod BA by 500 K.

TABLE 6. GAS RELEASE AFTER RAMPS FROM VILPPONEN[10]

Rod No.	Power (W/cm) Base	Final	Gas Release, % Measured	Calculated
105	289	631	17.8	18.4
109	239	642	12.1	20.0
110	256	594	26.3	23.2
2/7	430	605	29.7	26.5
2/8	430	606	32.1	26.5
3/7	430	564	35.7	25.5
3/8	430	558	26.9	26.5

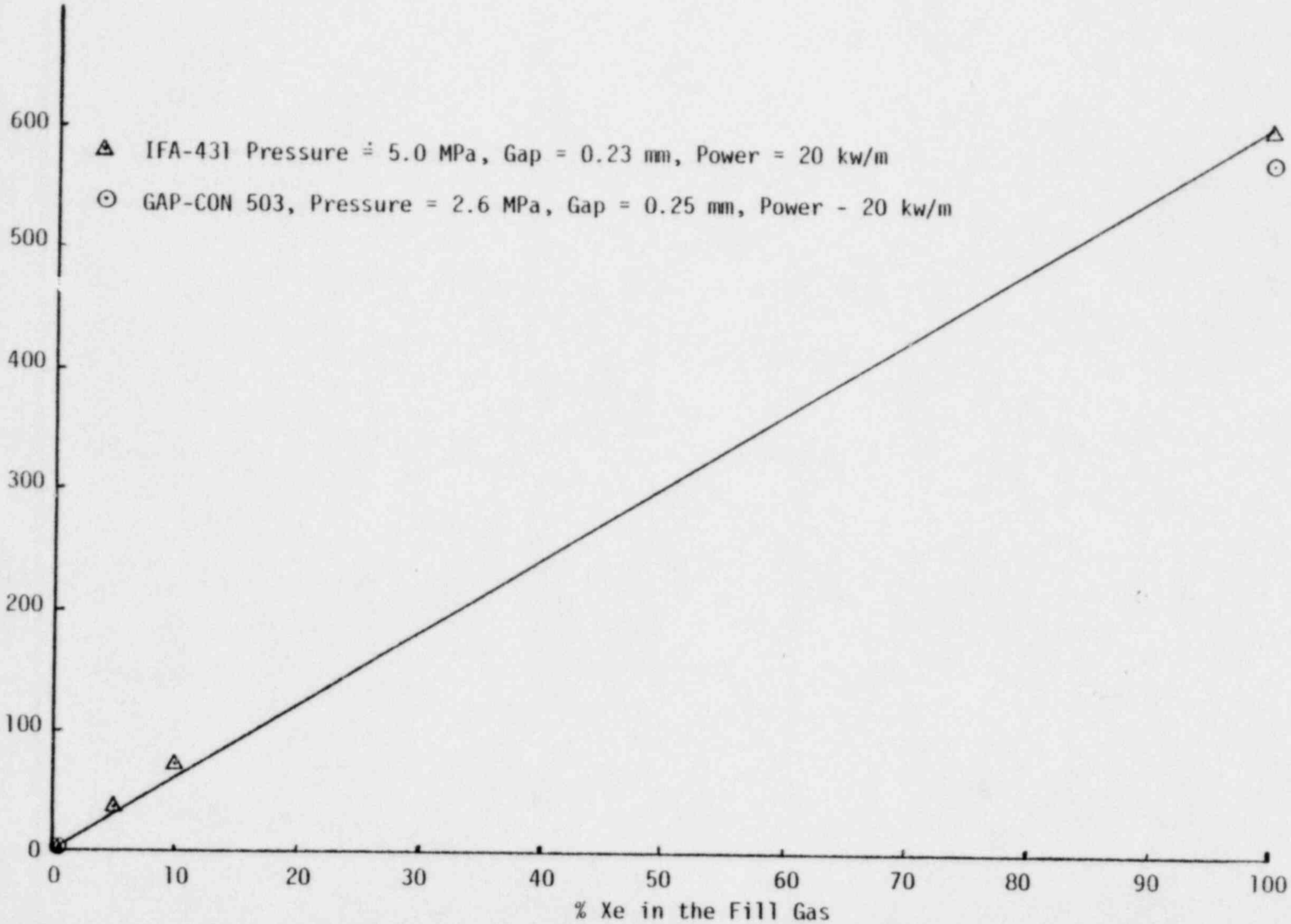

Figure 13. Fuel centerline temperature increase with fill gas composition.

As the pressure in Rod BA is unknown (it is suspected to have leaked into the TC lead) the amount of Xe that would have to have been released from the fuel to get a fill gas of 80% Xe/20% He cannot be directly determined. However, by assuming that the pressure in Rod BA is of the same order as that in Rod BB, the Xe release needed to give a 80/20 Xe to He ratio is estimated to be $\sim$2% of the total Xe produced.

In summary, the fission gas release predicted by various models is between 1 and 25% for Rod BA. The release required to raise the temperature of Rod BA to those measured depends on the pressure in Rod BA and, assuming that Rod BA leaked at the same rate as Rod BB, the release required can be calculated to be as low as 1%. Therefore it appears that the temperature rise measured in Rod BA could be due to fission gas release. This will be determined conclusively during the post irradiation examination of Rod BA that will be performed in late 1980.

3. Fuel Thermal Analysis

Thermal behavior data are available for Rods BA and BC. The rod BA thermal behavior was discussed in the previous section with respect to fission gas release. The thermal behavior of Rod BC has been uneventful, the fuel centerline temperature has increased by $\sim$8%, as shown in Figure 14, in the last two years. This temperature increase may be due to a small release of fission gas. None of the over-power transients (shield raisings) appear to have had any significant effect on fission gas release or thermal behavior of Rod BC.

IV. CONCLUSIONS AND RECOMMENDATIONS

The IFA-429 was designed to provide data on fill gas absorption, fission gas release, and fuel thermal behavior.

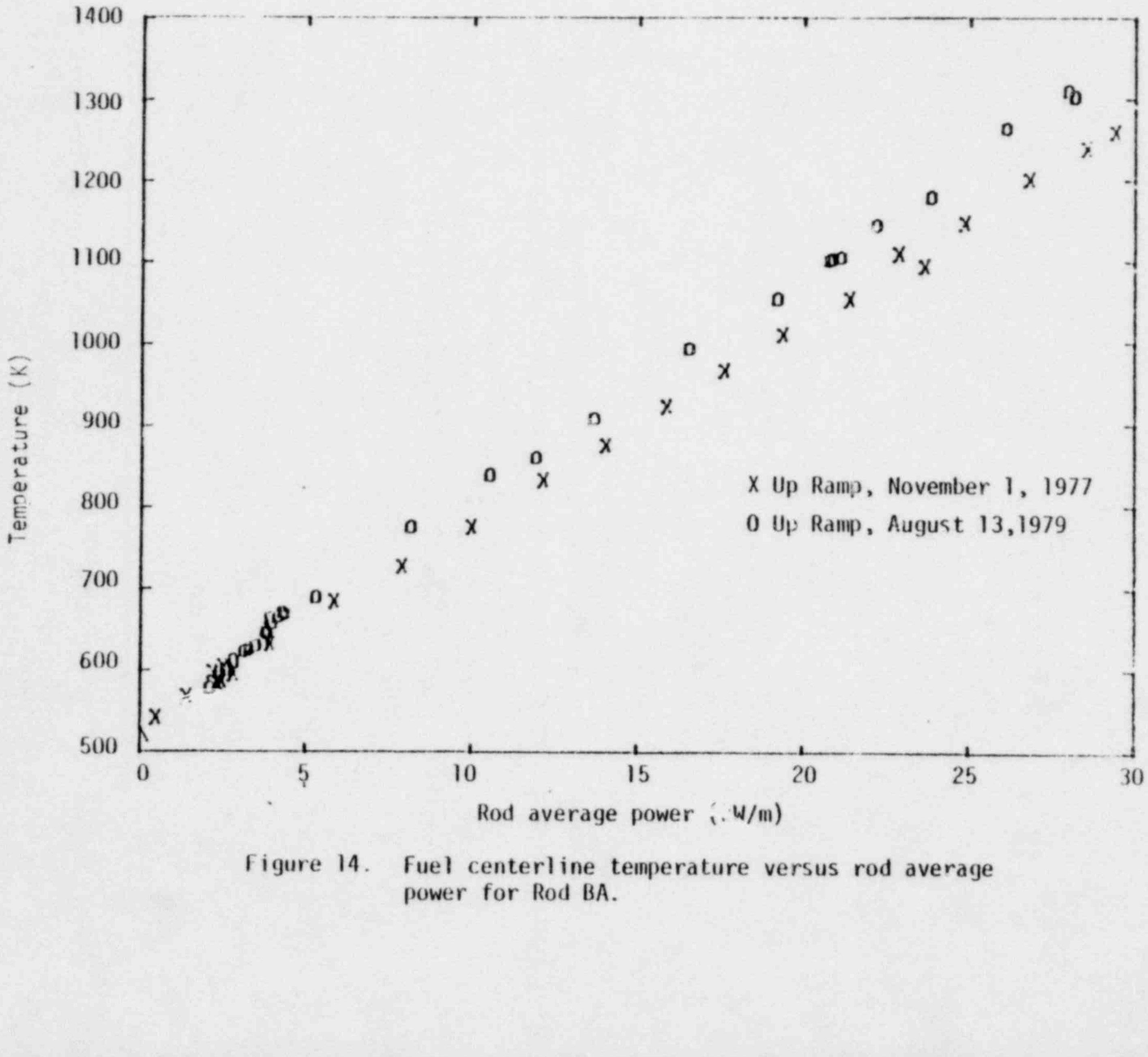

Figure 14. Fuel centerline temperature versus rod average power for Rod BA.

The absorption of the He fill gas has been measured and reported [1], the conclusions being that helium fill gas is absorbed into the UO_2 fuel in pressurized fuel rods. The data indicate that helium probably diffuses preferentially along grain boundaries or other short circuit paths and that helium absorption increases with fuel rod power. The absence of any measurable release of helium during high temperature (1950 K) transients indicates the absorbed helium is located at stable sites in the fuel. The amount of helium absorbed during the early stages of irradiation does not result in a rod pressure decrease sufficient to significantly alter the fuel rod behavior.

The release of fission gases in the fuel rods that have maintained the fabricated pressure (2.58 MPa, cold) has been insignificant, to the degree that it has not resulted in a statistically significant increase in fuel rod internal pressure. In the fuel rods whose internal pressures are low (1.5 MPA, hot) due to leakage through the instrument leads, some pressure data indicate fission gas release (increase in pressure) while others show no measurable release. However, as these rods do leak, it may be that as fission gas is released the rod internal pressure tends to equalize by forcing more gas into the instrument leads. Thus, any increase in pressure in the rods that are at low pressure (due to leakage) may indicate fission gas release but the absence of a pressure increase does not necessarily indicate that there has been no fission gas release.

The temperature response of the fuel, as a function of fuel rod power, can also indicate fission gas release because Xe released to the fuel cladding gap will degrade the gap conductance and result in increased fuel temperatures. Such a release is suspected in Rod BA; the release is estimated to have been 1 to 5% to result in the measured fuel temperatures. A release fraction of 1 to 25% is predicted by current steady state fission gas release models, indicating that such a release could be expected to occur. However, the uncertainty in the rod internal pressure prevents direct calculation of the release fraction at present. The rod in which the release is suspected to have occurred is to undergo postirradiation examination in late 1980, at which time the rod pressure and fission gas release fraction will be determined.

The only other rod with a fuel thermocouple has shown very little change in its thermal behavior since beginning of life. The temperature at the fuel centerline has increased by $\sim$ 8% indicating that there may be a small amount of fission gas release occurring.

The following are recommendations for the future test performance and analyses of IFA-429:

1. That the assembly continue operation to high burnup (> 50,000 MWd/t in the middle fuel cluster, > 30,000 MWd/t in the upper and lower fuel clusters) to determine when significant fission gas release will occur, if at all.

2. That the post irradiation examination of Rod BA specifically concentrate on fission gas inventory in the fill gas and retained in the fuel and that these data be combined with the fuel temperature data to define characteristics of the the fission gas release.

3. That the thermal behavior of Rod BC be closely monitored following overpower ramps to watch for indication of fission gas release.

4. That the ultrasonic thermometer data be evaluated to determine if they can be used as an indicator of fission gas release.

5. That a detailed analysis of the expected fission gas release be performed using the FRAP-T and GRASS-SST codes, and that the PIE data on fission gas release and retained fission gas be compared with GRASS-SST calculations.

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APPENDIX A

A review of eleven steady-state fission gas release models by Kolstad[a] is summarized below. The intention is to provide a brief, comparative, consistent evaluation, and to show how the different models correlate the fission gas release from UO_2 to local temperature and time (burn-up). Predicted fission gas release fractions, by these eleven models, for typical PWR and BWR fuel rods up to powers of 60 kW/m have shown considerable spread.

The models reviewed can be grouped into two categories depending on their dependence upon local fuel temperature. The first, the multi-stage models, assign constant maximum fission gas release to certain fuel temperature regions. For example, a four-stage model divides the fuel volume into four different temperature regions. The multi-stage model is based on the assumption that the liberation of fission gases is associated with fuel structural changes (viz. equiaxed growth, columnar grain growth etc.) which are observed to be temperature dependent.

The second category, the diffusion-based models, predict the fission gas release to increase gradually with local temperature (no abrupt changes occur at any temperature). This approach is based on the assumption that the release is controlled by a temperature dependent diffusion mechanism. The effective diffusion constants and activation energies may vary significantly in different temperature regions of the fuel.

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a. E. Kolstad, A Review and Comparative Evaluation of Eleven UO_2 Fission Gas Release Models, Enlarged Halden Programme Group Meeting on Fuel Performance Experiment and Evaluation, Hanko, Norway, June 17-21, 1979. Volume II, August 1979.

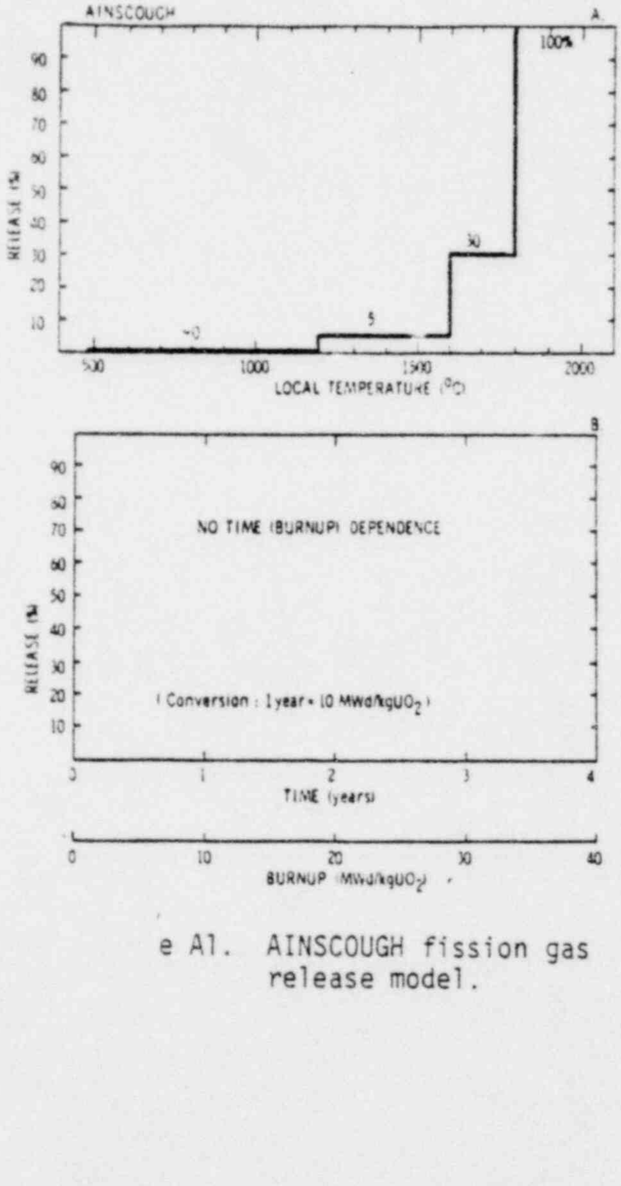
Both types of models can incorporate the burnup effect on release. The burnup effect could even be an indirect one as in the case of a prescribed incubation period for the onset of release. In all models the release rate, for a given fuel temperature condition, increases with burnup. However the rate of increase is higher at low and intermediate burnups and temperatures. The burnup effect on fission gas release, at high exposure, becomes smaller and tends to saturate. Figures A-1 through A-11 show the dependence of the 11 fission gas release models reviewed (Table A-1) on temperature and burnup .

TABLE A-1. MODEL COMPARISON. (TEMPERATURE/TIME/BURNUP DEPENDENCES)

Correlation No.	Multistage Model No. of Stages	Continuous Temp. Dependence	Burn-up[3] Dependence
1. Ainscough	4		No
2. Beyer-Hann (original)	4		No
3. Beyer-Hann (revised)	4		No
4. LOOPY	4		No
5. BFNPA	3 (2)[1]		Yes[4]
6. Lewis	4		Yes[5]
7. CE	4		Yes[6]
8. Vitanza		x	Yes[7]
9. Joon		x[2]	No
10. Diffusion		x	Yes[8]
11. FRAP-S		x	Yes[8]

Notes:

1. 3 stages at burnup <15000 MWd/t, 2 at burnup >15000 MWd/t.
2. Release is correlated to fuel center temperature only.
3. The time (burnup) dependence decreases with time.
4. Two versions depending upon whether Bu $\lessgtr$ 15000 MWd/t.
5. Release is proportional to $\sqrt{t/3}$ (t = time, year) for temps < 1600°C.
6. As 5., but for temps < 1840°C.
7. Indirect burnup dependence (incubation period for fission gas release).
8. Release is time (burnup) dependent at all temperatures.



e A1. AINSCOUGH fission gas release model.

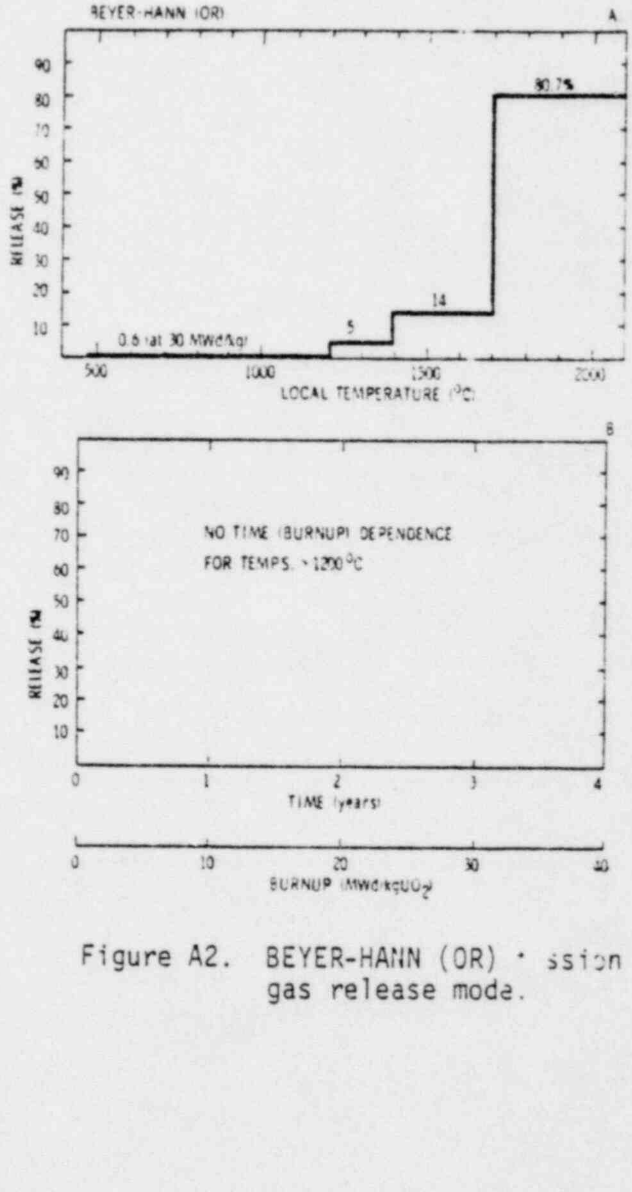


Figure A2. BEYER-HANN (OR) fission gas release mode.

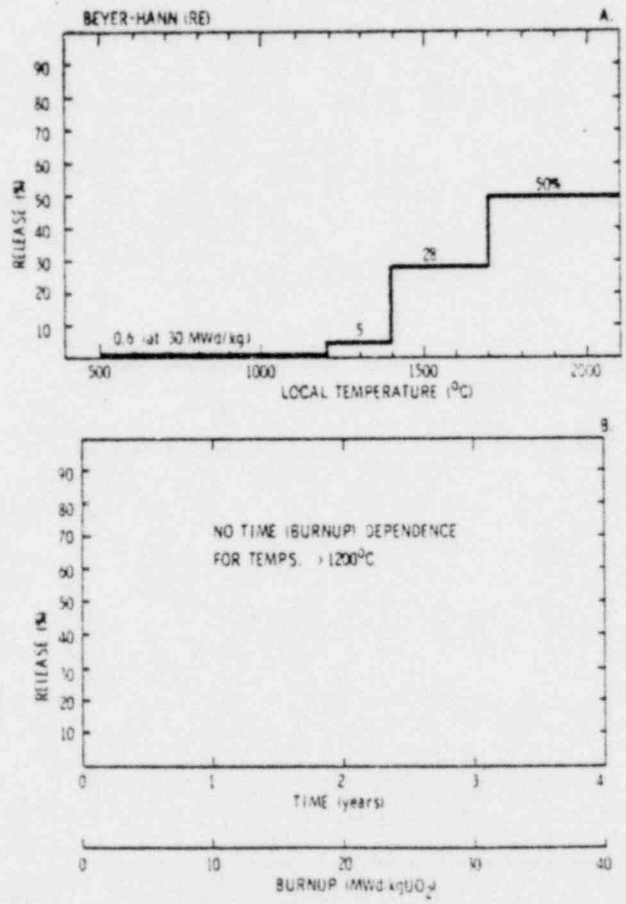


Figure A3. BEYER-HANN (REVISED) fission gas release model.

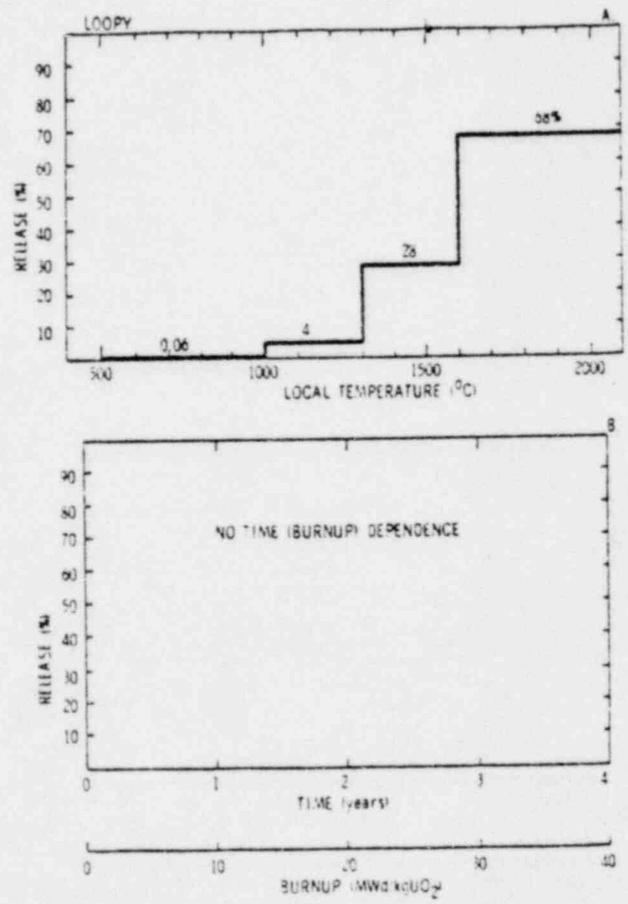


Figure A4. Loopy fission gas release model.

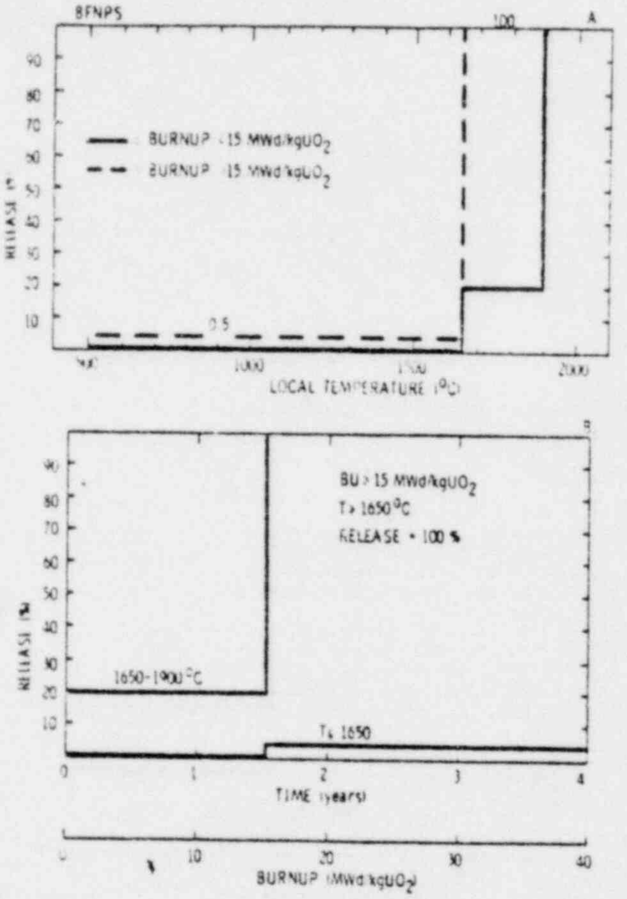


Figure A5. BFNPS fission gas release model.

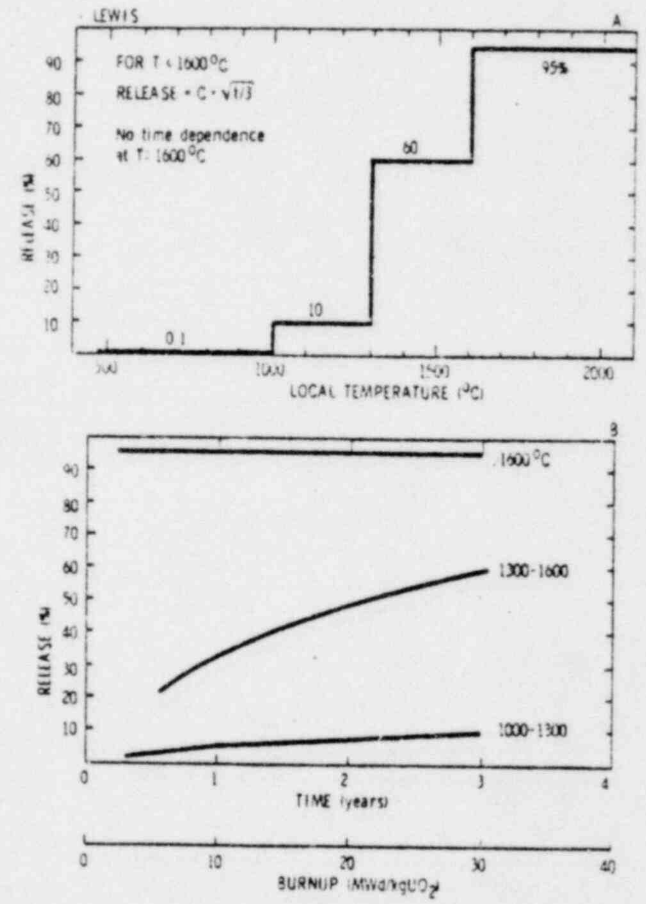


Figure A6. LEWIS fission gas release model.

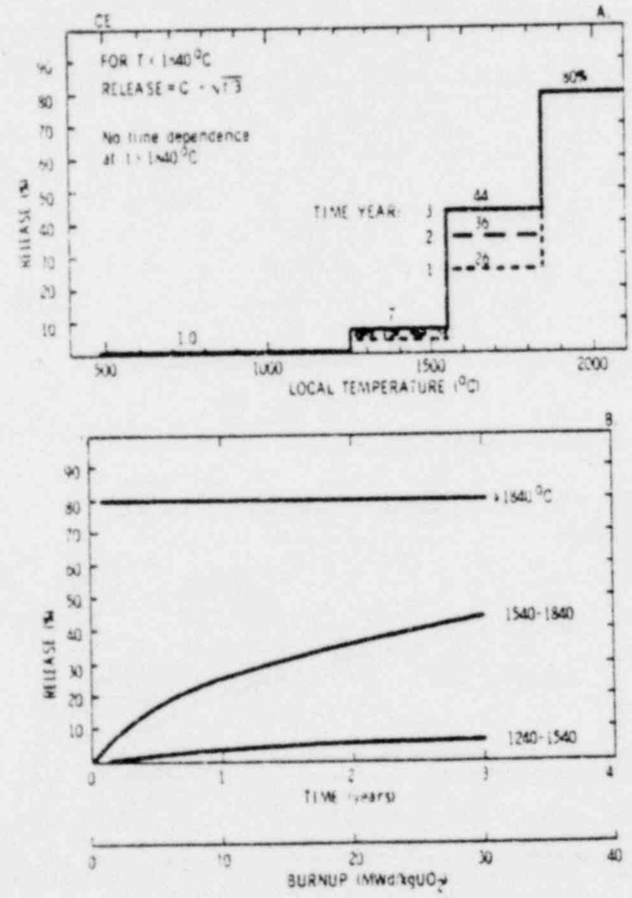


Figure A7. CE fission gas release model.

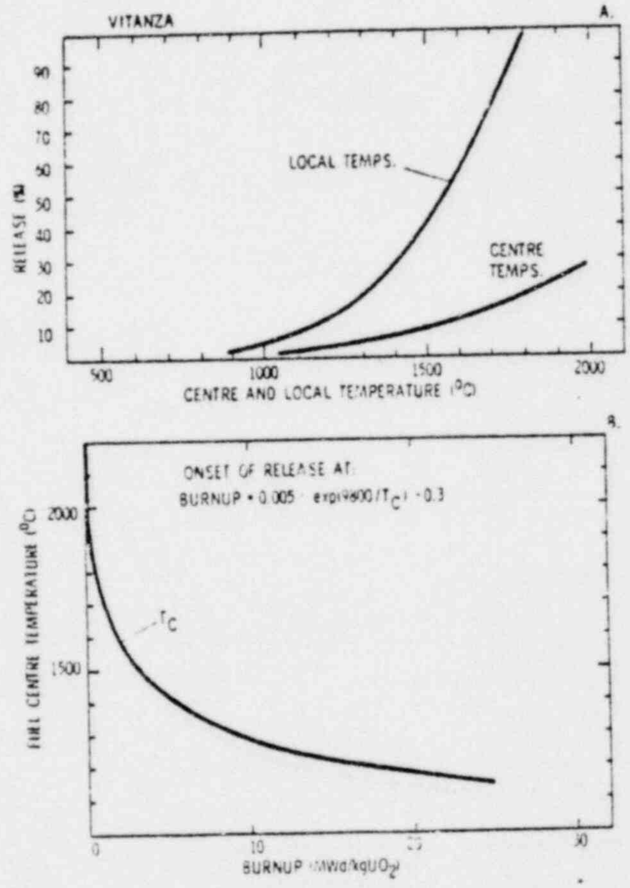


Figure A8. Vitanza fission gas release model.

Vitanza

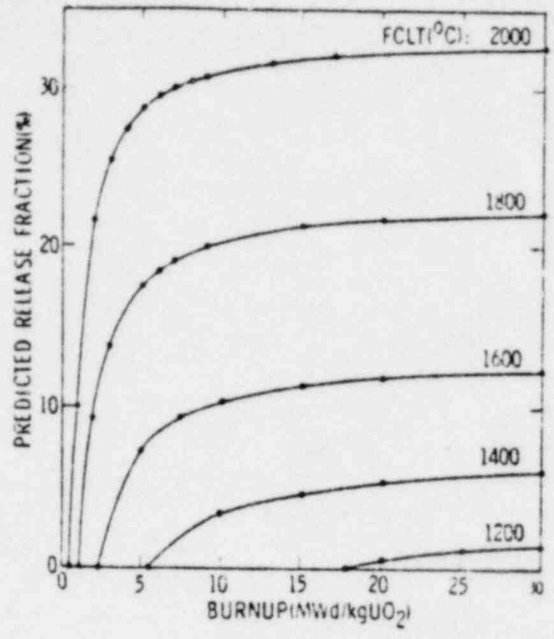


Figure A9A

Vitanza

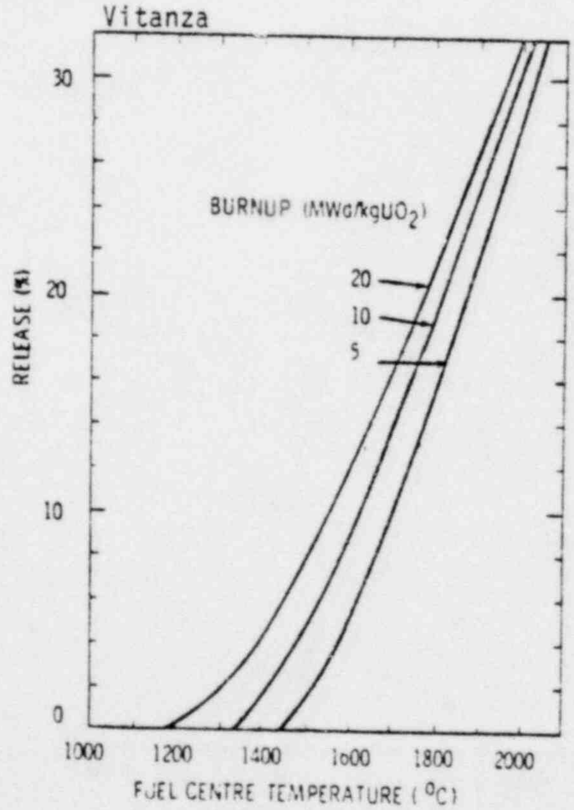


Figure A9B

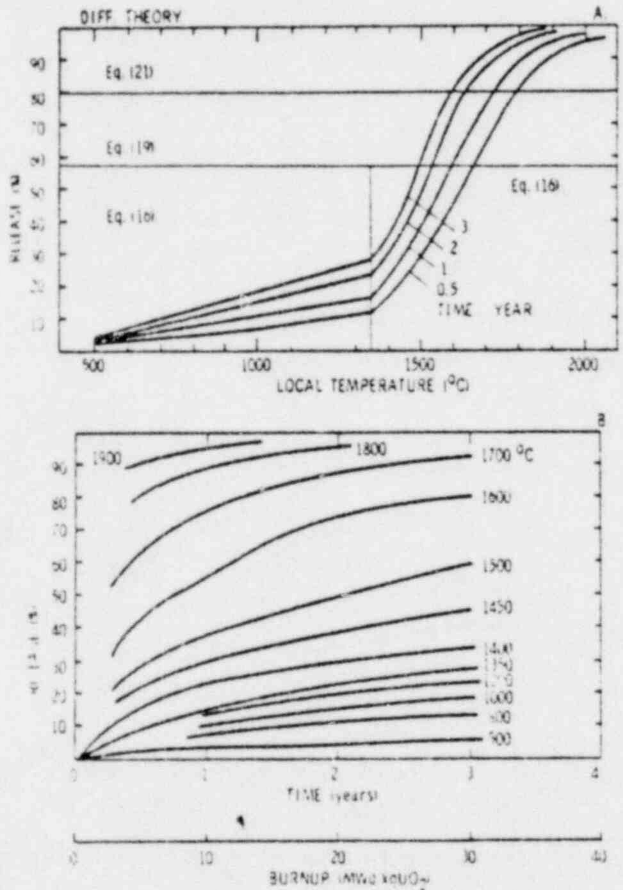


Figure A10. Diffusion theory fission gas release model.

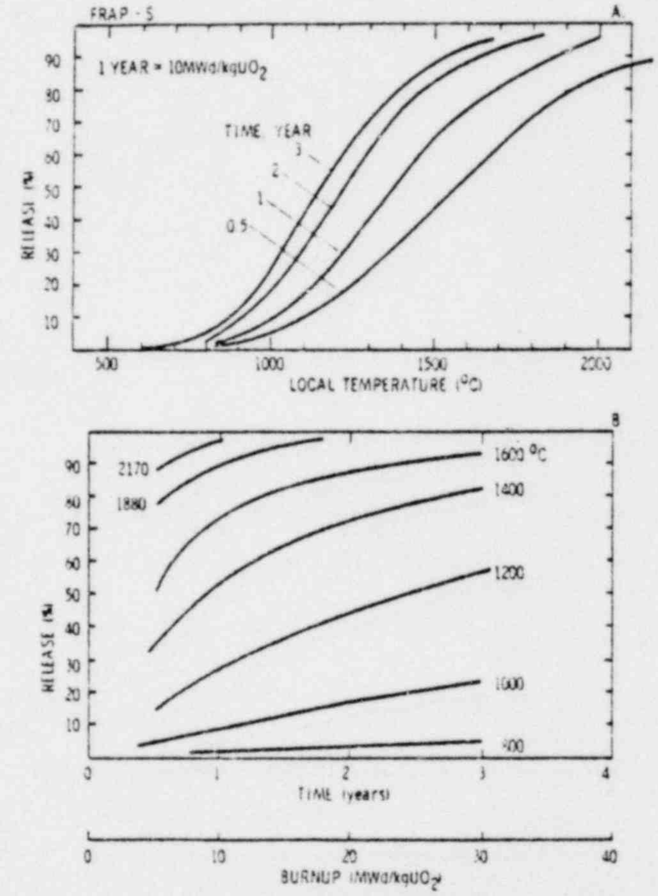


Figure A11. FRAP-S fission gas release model.