END OF CYCLE BURN UP'S

Total energy retrieved 38150 MWd

	A	B	C	D
4		P1 23100	P2 22300	
3	F3 13000	F1 17000	C2 27600	C3 22600
2	C4 23700	C1 28600	F2 16600	P4 22100
1		F4 12600	P8 23300	

Fig.6

LWR FUEL PERFORMANCE DURING ANTICIPATED TRANSIENTS WITH SCRAM*

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ABSTRACT

Operational transients occur occasionally in light water reactors when minor malfunctions of certain system components affect the reactor core. Potential effects of such malfunctions include a loss of the secondary heat sink, an increase in system pressure, and, in boiling water reactors, void collapse and a brief increase in reactor power. The most severe postulated Boiling Water Reactor (BWR) anticipated transient is characterized by a power peak of up to 495% rated power for about 1 second (according to a recent General Electric Co., generic analysis). The results of a series of fuel behavior tests in the Power Burst Facility (PBF) at the Idaho National Engineering Laboratory are presented in this paper. Four progressively higher and broader power transients at a constant coolant flow rate were performed. The first transient simulated a BWR-5 turbine trip without steam bypass with fuel rods operating at BWR-6 core average rod powers. The second transient simulated a generator load rejection without steam bypass with fuel rods operating at above core average powers. The last two transients were performed at higher powers than safety analysis predicts to be possible in commercial reactors to define failure threshold margins. The test rods did not fail and were not damaged during any of the four transients.

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INTRODUCTION

Anticipated nuclear power reactor transients are deviations from normal plant operating conditions that result from system component malfunctions or reactor operator errors which may occur one or more times during the service life of a reactor and are normally accompanied by a control rod scram. They are distinguished from "accidents," which have a much lower probability of occurrence and may result in much more severe consequences. The Electric Power Research Institute (EPRI) has selected 37 categories of anticipated and unanticipated BWR malfunctions and 41 categories of pressurized water reactor (PWR) malfunctions on the basis of transients defined in the NRC assessment of accident risks in U.S. nuclear power plants[1] and data from utilities for transients that have actually occurred.[2] These transients have been assigned a frequency of occurrence per reactor year from 0.02 ± 0.14 to 1.41 ± 1.89 for BWRs and from 0.01 ± 0.09 to 1.69 ± 2.44 for PWRs.

The effects of such malfunctions may include a loss of the secondary heat sink, an increase in system pressure, and, in boiling water reactors, void collapse and a brief increase in reactor power. The most severe postulated BWR-5 anticipated transient is a generator load rejection without steam bypass, which is characterized by a peak transient power spike of up to 495% of rated power for about 1 second. Dryout and severe cladding temperature excursions are not expected during such transients and, therefore, the damage mechanism of concern is cladding fracture due to pellet-cladding mechanical and chemical interactions.

The first indication that zircaloy-clad UO_2 rods might be susceptible to failure due to a pellet-cladding interactive mechanism inherent to the fuel and cladding materials was obtained in 1964 by the General Electric Co., in the "High Performance UO_2 Program," jointly sponsored by the United States Atomic Energy Commission and EURATOM.[3] Since that time, the phenomena of pellet-cladding interaction (PCI) induced cladding failure during normal light water reactor operation have received considerable attention throughout the world. Such failures are apparently induced by power increases after a sufficiently high burnup is attained to allow fission product release. There has been a strong incentive to find a remedy for these failures because the present method of preventing such failures is to accept limits on rates of reactor power increase. These limits are expensive due to the lost power output during slow increases. Experiments have been performed in the Halden, Studsvik, NRU, GETR, RISO, RCN-Petten, BR-2 and BR-3 reactors.[4-9] Most investigators now accept the view that both the presence of aggressive chemical species and high localized stresses are prerequisites for power ramp induced pellet-cladding interaction failures.[10] However, pellet-cladding mechanical interaction failures have also occurred during severe power increases due to high strain rate tearing or fracture of irradiation-embrittled zircaloy cladding.[11]

Since severe core power increases are possible during a variety of anticipated transients and the most severe postulated anticipated transients have not yet occurred in commercial reactors, the U.S. Nuclear Regulatory Commission (USNRC) was uncertain whether light water reactor fuel rods would fail or can be damaged during such events. Therefore, a series of in-pile fuel behavior tests labeled OPT I-1 were conducted in the Power Burst Facility (PBF) by EG&G Idaho, Inc., for the USNRC to (a) determine the threshold at which light water reactor fuel rods are likely to fail during severe anticipated transients which result in a brief increase in reactor power and (b) identify any fuel and damage mechanisms which may occur. The PBF data, along with other test data, will be used by the USNRC to assess the failure probabilities used in licensee dose calculations for anticipated transients. These results may also impact other questions such as: (a) should a reactor be derated following a severe anticipated transient, (b) should PCI-damaged fuel be removed following a transient, and (c) should regulations be imposed to limit pellet-cladding interaction in irradiated fuel rods?

TEST CONDUCT

Six fuel rods originally fabricated by the General Electric Co., and irradiated in the Northern States Power Company's Monticello boiling water reactor to burnups ranging from about 5,000 to 23,000 MWd/t were tested. Four of the six fuel rods were typical BWR-6 design rods, except for fuel length (0.75 m) and plenum volume (which was scaled to the fuel length). Two of the fuel rods incorporated design modifications to improve their PCI resistance. Each fuel rod was surrounded by an individual flow shroud and four fuel rod and shroud assemblies were symmetrically placed within the PBF, as shown in Fig. 1, for each transient.

Each fuel rod was fixed rigidly to the shroud at the top of the fuel rod and was free to expand axially downward against a linear variable differential transformer (LVDT) that measured the axial growth of each rod. Additional instrumentation was provided to measure coolant conditions, fuel rod power, and fission product release. The test rods were not opened prior to the PBF tests and contained no instrumentation.

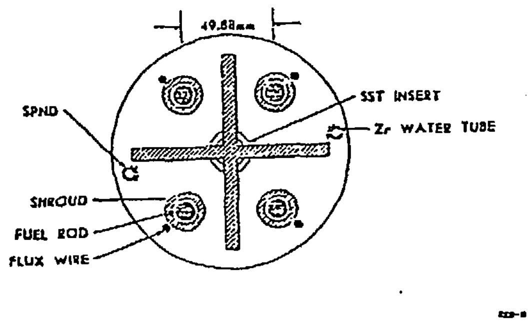

Figure 1. Schematic of PBF 4X test hardware.

The nuclear operation in the PBF consisted of two extensive fuel conditioning phases and four power transients. The purpose of the fuel conditioning was to measure the ratio of PBF core power to test rod power and to carefully condition the fuel rods to a peak rod power of 27 kW/m, since the test rods had been irradiated in the Monticello BWR at the edge of the core at a power of only about 13 kW/m. The conditioning consisted of a slow power ramp with single-phase coolant conditions to a rod power of ~27 kW/m. Maximum rod power ramp rates were held to 0.5 kW/m per minute up to 25 kW/m and 0.35 kW/m per hour from 25 to 27 kW/m. Each of the two fuel conditioning phases extended over approximately 28 hours.

The four progressively higher and broader power transients shown in Fig. 2 were conducted at power ramp rates as high as 550 kW/m per second. The power-time histories specified for the first two transients (Transients A and B) approximate the results of a conservative analysis of various BWR-5 anticipated transients performed by the General Electric Co., using the ODYN computer code. The last transient was conducted at the physical limits of the PBF. Approximately a 2-hour hold at steady power preceded each transient. The nominal coolant temperature, flow rate, and pressure conditions during each transient were 550 K, 525 cm^3/s, and 7.93 MPa, respectively.

The first transient simulated a BWR turbine trip without steam bypass, with the irradiated fuel rods operating at typical BWR core average powers (~26 kW/m). The peak fuel rod power was increased from 26 kW/m to 92 kW/m in 0.32 s while maintaining a constant coolant flow rate during the power transient. The PBF was able to almost exactly reproduce the specified power history, as illustrated in Fig. 3. Following the first transient, the loop was cooled and depressurized, the test train removed from the in-pile tube, and two of the standard BWR-6 fuel rods were removed and replaced with two other reference-type fuel rods. The two test rods were removed after the first transient so that the possible occurrence of incipient PCI cracks on the inside surface of the cladding could be investigated.

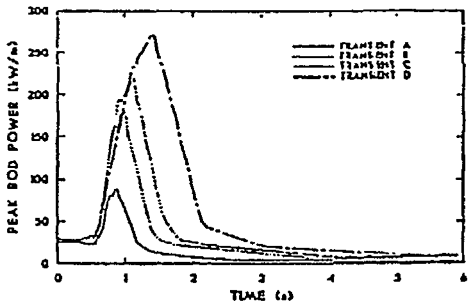

Figure 2. Peak rod power versus time specified for the OPT 1-1 transients.

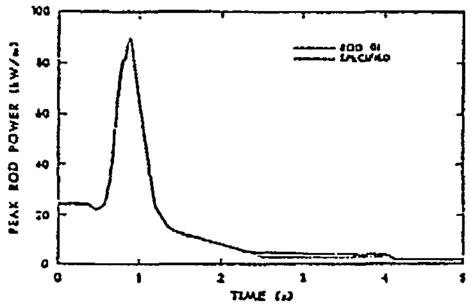

Figure 3. Comparison of measured and specified Rod 1 power during OPT -1- Transient A.

The power history of the second transient (28 to 177 kW/m in 0.66 s) simulated a BWR generator load rejection transient without steam bypass for fuel rods operating above BWR core-average rod powers, except that the time duration of the transient was about twice that predicted by the General Electric Co.

The third and fourth transients were performed at higher transient powers than current safety analysis predicts to be possible in an effort to determine failure threshold margins. The peak fuel rod power was increased from 28 to 206 kW/m in about 0.74 s during the third transient and from 28 to 261 kW/m in 0.96 s during the fourth transient.

TEST RESULTS

Table I summarizes the calculated fuel enthalpies and fuel temperatures reached during the transients. The radially averaged peak fuel enthalpy increased from 47 to 87 cal/g UO_2 during the fourth transient and the fuel centerline temperature increased from 1350 to 2005 K following the transient. A maximum cladding axial elongation change of 2.6 mm was measured during the fourth transient. Hard pellet-cladding contact was calculated to result in a maximum cladding hoop strain of 0.44% and a hoop stress of 183 MPa. As expected, boiling transition did not occur on any of the fuel rods.

TABLE I. OPT 1-1 POWER TRANSIENTS

Transient	Initial Fuel Rod Power (kW/m)	Initial Fuel Enthalpy (cal/g)	Peak Fuel Rod Power (kW/m)	Peak Fuel Enthalpy (cal/g)	Centerline Temperature (K)
A	26.0	45	92	49	1350
B	28.4	48	177	63	1590
C	28.5	48	206	69	1695
D	26.2	47	261	87	2005

Fission products were not released during or after any of the four power transients and posttest analysis of the plenum gases confirmed that none of the fuel rods leaked. In fact, we have not been able to observe any damage or change in these rods which could have been caused by the PBF testing. The results of the posttest plenum gas analysis are summarized in Table II. A cross section of Rod 3 (the highest burnup rod) at the peak power location is shown in Fig. 4. An enlargement of the Rod 3 cladding structure is shown in Fig. 5. The results summarized in Table II and shown in Figs. 4 and 5 are typical of normal, undamaged light water reactor fuel.

TABLE II. RESULTS OF THE POSTTEST PLENUM GAS ANALYSIS

Rod	Total Gas in STD cm^3 [a]	Void Area in cm^3 [b]	MOL (%) He	MOL (%) Kr	MOL (%) Xe	Rod Type	Average Burnup (GWd/t)	Original BWR Axial Location[c]
1	13.4	14.4	97.5	0.24	1.83	Reference	13.5	Bottom
2	37.9	13.8	99.3	0.07	0.51	Zirconium liner	5.0	Lower Middle
3	13.4	13.8	96.6	0.33	2.79	Reference	22.8	Bottom
4[d]	35.1	14.1	97.4	0.15	0.77	Fuel additive[e]	5.1	Lower Middle
5	13.3	13.7	94.6	0.53	4.14	Reference	12.1	Bottom
6	12.8	13.3	94.7	0.44	3.40	Reference	15.4	Bottom

a. ±0.2 accuracy.

b. ±0.1 to 0.3 accuracy.

c. Segmented rods were irradiated in a bundle located on the extreme periphery of the Monticello BWR owned and operated by Northern States Power Co. Four segmented rods were fastened together to form a ~3.86-m long rod.

d. Small water leak into sample.

e. Composition of fuel additive rod is proprietary General Electric Co. information. Measurements by General Electric Co. indicate that the conductivity and thermal expansion of a fuel additive rod are unchanged relative to UO_2. The fuel melting point of a fuel additive rod is estimated to be 70 K lower than for UO_2.

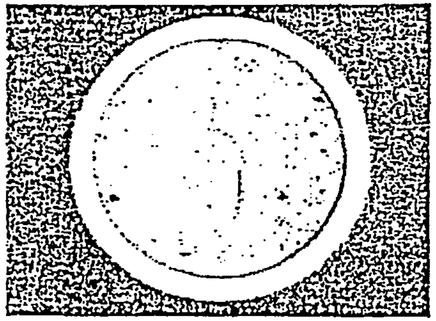

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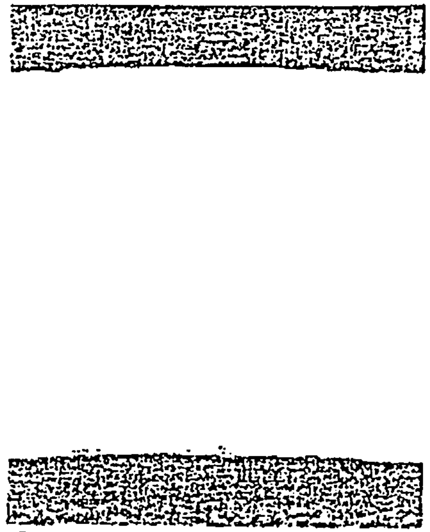
Figure 4. Cross-section of Rod 901-3.



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100 μm

Figure 5. Typical cladding structure of Rod 901-3.

CONCLUSIONS

Even though only six fuel rods were tested during the OPT 1-1 Test Series, the peak transient fuel rod powers were twice that expected for a design average power rod (26 kW/m) subjected to the worst anticipated transient presently considered credible for a BWR, and none of the test rods failed. Although postirradiation examinations are continuing, the lack of any evidence of cladding through-wall cracks strongly suggests that BWR fuel rods will not fail during brief power transients. The severity of the tests compensates somewhat for the lack of redundancy in test rods with regard to possible interpetation of the significance of these results. However, the fuel rods used in the OPT 1-1 tests and the PBF test conditions are not entirely typical of those in commercial reactors. For instance, the short rod length may have affected the fission product release and transport and the axial loading of the cladding due to pellet cladding mechanical interactions. Therefore, further evaluation of the question may be required.

XA0201781

NOTICE

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COMPARATIVE STEADY-STATE AND POWER RAMPING PERFORMANCE OF ANNULAR-COATED-PRESSURIZED, SPHERE-PAC AND REFERENCE TEST RODS IN THE HALDEN BWR

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1.0 INTRODUCTION

The Fuel Performance Improvement Program (FPIP) is sponsored by the U.S. Department of Energy (DOE) and is performed by Consumers Power Company (CPC), Exxon Nuclear Company, Inc. (ENC), and the Battelle Pacific Northwest Laboratory (PNL). One objective of the FPIP is to better understand the pellet-cladding interaction (PCI) behavior of advanced LWR fuel designs during steady-state and power-ramping operation. The described work summarizes that part of the FPIP done at the Halden Boiling Water Reactor (HBWR).

2.0 DESCRIPTION OF THE EXPERIMENTS

Fuel rods of advanced design were irradiated in instrumented fuel assembly test rigs to burnup levels up to 16 MWd/kgM at linear heat generation rates (LHGR) of up to ~40 kW/m. These base irradiations were followed by power ramps to terminal powers in the range from 66 to 74 kW/m.

The following fuel designs were studied in these instrumented test rigs:

A reference design (R) of solid pellets with dished ends equivalent to 1.0 vol% of an undished pellet. The reference design was pressurized to 0.10 MPa (1.0 atm) helium.