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IDENTIFICATION OF THE SECURITY THRESHOLD BY LOGISTIC REGRESSION APPLIED TO FUEL UNDER ACCIDENT CONDITIONS

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ABSTRACT

A reactivity-initiated Accident (RIA) is a disastrous failure, which occurs because of an unexpected rise in the fission rate and reactor power. This sudden increase in the reactor power may activate processes that might lead to the failure of fuel cladding. In severe accidents, a disruption of fuel and core melting can occur. The purpose of the present research is to study the patterns of such accidents using exploratory data analysis techniques. A study based on applied statistics was used for simulations. Then, we chose peak enthalpy, pulse width, burnup, fission gas release, and the oxidation of zirconium as input parameters and set the safety boundary conditions. This new approach includes the logistic regression. With this, the present research aims also to develop the ability to identify the conditions and the probability of failures. Zirconium-based alloys fabricating the cladding of the fuel rod elements with niobium 1% were analyzed for high burnup limits at 65 MWd/kgU. The data based on six decades of investigations from experimental programs. In test, perform in American reactors such as the transient reactor test (TREAT), and power Burst Facility (PBF). In experiments realized in Japanese program at nuclear in the safety research reactor (NSRR), and in Kazakhstan as impulse graphite reactor (IGR). The database obtained from the tests and served as a support for our study.

1. INTRODUCTION

The objective of this investigation is to build a prediction model based on a statistical method, the logistic regression (LR) theory. Currently, studies on nuclear fuels seek to extend irradiation cycles for economic reasons. Researchers have focused on replacing zirconium alloys with advanced alloys. Innovative alloys could continue the irradiation cycles for a long time under stringent safety conditions. The global effort in the field of nuclear fuels focuses on the properties of cooling environments, manufacturing processes in the absence of defective fuels, and the improvement over the design of fuel assembly. The statistical fitting was the outcome of the application of data analysis to large samples. The LR can detect the risk of failure. The logistic model is as good as an ordinary regression model. However, a binary variable is the result of LR.

This type of variable assumes two values for accident conditions, such as failure (1) or success (0). The identification of the safety threshold for an RIA accident requires full knowledge about nuclear fuel under transient conditions. The objective of this examination is to determine a statistical correlation between enthalpy and burnup for predicting the security threshold for light-water reactor (LWR) fuels. Currently, several studies focus on the nuclear fuel for extending irradiation cycles.

2. BACKGROUND

A reactivity accident is a disastrous failure, which occurs because of an unexpected increase in the fission rate and reactor power. This sudden increase of power may activate processes that might lead to the failure of the fuel. In severe accidents, the disruption of fuel rod and core melting can occur. The purpose of the present research is to study an empirical correlation-based statistical approach. During long cycles, a rapid degradation of materials occurs because of a high flux of neutrons. One of the most common causes of failure includes cladding corrosion and hydride deposition. The stress and strains in cladding are induced by pellet-cladding mechanical interaction (PCMI) [1].

2.1. Logistic Regression

The statistical regression method is as good as an ordinary regression model. However, the result is a binary variable; this variable assumes two values, such as a failure or success, for accident conditions. The LR is a regression analysis where the dependent variable is a dummy variable (coded 0, 1) [2]. The LR methodology or a logit model is the log odds of the outcome is a combination of predictor variables. The model is simply a non-linear transformation of the linear regression [3].

2.2. Control Rod Ejection Accident

The typical scenarios postulated for reactivity insertion are control rod ejection accident (CREA) in a pressurized water reactor (PWR), or control rod drop accident (CRDA) in a boiling water reactor (BWR) [4]. A control rod element consists of a stainless steel cladding containing B4C pellets with enriched boron as the neutron absorber. During the start-up phase, the reactor is heated from cold zero power (CZP) to hot zero power (HZP). At CZP, all the control rods inserted sequentially in the core are slowly withdrawn [5].

2.3. Safety Experiments

Safety tests for nuclear fuel were conducted for the first time around 1950. In 1959, an individual reactor called transient reactor test facility (TREAT) was built to investigate fuel melting [6]. In 1972, experimental programs were performed in power burst facilities. The PBF operated from 1972 to 1985 to study the performance of LWR fuel elements during transients. The first unit of a special power excursion reactor test (SPERT) became operational in July. This test unit could simulate many fuel rods during a transient analysis. A pulse reactor was built in Kazakhstan. In a fast-pulse graphite reactor (BIGR), 12 core safety tests were performed. The nuclear safety research reactor is an adapted TRIGA reactor so called as annular core pulse reactor (ACPR) [7].

The pulse reactor became operational in 1975 with the intention of conducting research on fuel behavior, mostly under RIA conditions [8]. A CABRI reactor localized at Cadarache, France consisted of a water pool reactor with an experimental sodium loop. The fast breeder reactor (FBR) program was conducted from 1978 to 2001 with cooled sodium. The LWR program was carried out from 1993 to 2002, in which UO_2, $(Pu\text{-}U)O_2$, and irradiated fuels were tested. Currently, thirty countries contribute to the CABRI international program (CIP) [9]. The NSRR conducted over thousand safety tests for PWR, BWR, and mixed oxide fuel (MOX) [10]. The experiments were performed on the NSRR and CABRI considering the future burnup increase in commercial reactors [11].

3. MATERIAL AND METHODS

The database has 550 samples; a series of the experimental rods formed about 150 cases and a sequence of 400 extra cases producing a significant database.

3.1. Reactivity Initiated Accident Benchmark

The PBF in Scoville, US, conducted experiments from 1978 to 1980 simulating an RIA, led by Idaho national laboratory (INL). The BIGR in Kazakhstan proposed to investigate the effects of a narrow pulse from 2.5 to 3.1 ms and the loss of ductility for (Zr-1%Nb) cladding. The experiments conducted at PBF are listed in Table 1. The pulse width ranges from 2 to 3 ms; the fuel rod has an active length of 150 mm. The experiments conducted at BIGR are listed in Table 2.

Table 1: PBF tests PWR fuel

Rod	Bu(MWd/kgU)	Oxide(um)	Enthalpy (cal/g)	Pulse Width (ms)	Fail
801-1	4.6	5	285	13	1
801-2	4.7	5	285	13	1
801-3	0.0	0	285	13	1
801-4	0.0	0	285	13	1
802-1	5.2	5	185	16	0
802-2	5.1	5	185	16	0
802-3	4.4	5	185	16	1
802-4	4.5	5	185	16	0
804-1	6.1	5	185	11	0
804-3	5.5	5	185	11	0
804-4	5.0	5	255	11	0
804-5	5.5	5	234	11	0
804-6	5.1	5	255	11	0
804-7	5.9	5	253	11	0
804-8	5.9	5	255	11	0
804-9	5.7	5	253	11	0
804-10	4.4	5	255	11	0

The pulse widths used are from 2 to 3 ms; with the fuel rods having an active length of 150 mm. The current investigation programs show in the course the Cabri International Program (CIP) with the contribution of 30 countries for high burnup fuel studies . The NSRR produced over thousand safety tests of PWR, BWR, and Mixed Oxide fuel, or MOX. The experiments performed on NSRR and CABRI program view with the future extended cycles of the irradiation increase in the commercial reactors [11]. Pulse Graphite Reactor (BIGR) in Kazakhstan. The BIGR program proposed to investigate effects of a narrow pulse from 2.5 to 3.1 ms, loss of ductility for (Zr-1%Nb) cladding. It illustrated in Table 2, where we show the experiments conducted at BIGR.

Table 2: BIGR tests VVER fuel

Rod	Bu (MWd/KgU)	Oxide (um)	Enthalpy (cal/g)	Pulse Width (ms)	Fail
RT1	48	5	142	2.6	0
RT2	48	5	115	3.1	0
RT3	48	5	138	2.5	0
RT4	60	5	125	2.5	0
RT5	49	5	146	2.5	0
RT6	48	5	153	2.6	0
RT7	60	5	134	2.6	0
RT8	60	5	164	2.6	1
RT9	60	5	165	2.7	1
RT10	47	5	164	2.6	1
RT11	49	5	188	2.6	1
RT12	47	5	155	2.8	0

The experiments were conducted in the special power excursion reactor test - capsule driver core (SPERT-CDC) from 1969 to 1970. The average burnups were in the range of 1–32 MWd/kgU. The experiments conducted in SPERT-CDC are listed in Table 3.

Table 3: SPERT-CDC tests BWR fuel

Rod	Bu(MWd/kgU)	Oxide(um)	Enthalpy (cal/g)	Pulse Width (ms)	Fail
CDC-567	3.1	0	214	18	1
CDC-568	3.8	0	161	24	1
CDC-569	4.1	0	282	14	1
CDC-571	4.6	0	137	31	0
CDC-684	13	0	170	20	0
CDC-685	13	0	158	23	0
CDC-703	1.1	0	163	15	0
CDC-709	1.0	0	202	13	1
CDC-756	32	65	143	17	1
CDC-859	32	65	154	16	1

Experiments were conducted in the CABRI reactor, located at Cadarache research in southern France, from 1993 to 2002. Pulse widths of 9–75 ms were applied during the tests performed at CABRI resulting in a burnup of 22–77 MWd/kgU and cladding layer oxides of 4–100 µm. The CABRI water loop operated from 2000 to 2015. The project is investigating the ability of the nuclear fuel to withstand the increase in reactor power within few milliseconds. Table 4 lists the experiments conducted in CABRI program. Table 5 lists the experiments conducted at NSRR for PWR fuels. Table 6 lists the tests done at NSRR for BWR fuels. The Japanese project in NSRR is an in-pile testing program using a pulse reactor.

Table 4: The CABRI tests with PWR fuel

Test	Bu (MWd/KgU)	Oxide (µm)	Enthalpy (cal/g)	Pulse Width (ms)	Fail
NA1	64	80	30	9.5	1
NA2	33	4	199	9.5	0
NA3	54	40	124	9.5	0
NA4	62	80	95	76	0
NA5	64	20	108	8.8	0
NA6	47	40	133	32	0
NA7	55	50	114	40	1
NA8	60	130	82	75	1
NA9	28	20	197	33	0
NA10	63	80	80	31	0
NA11	63	15	93	31	0
NA12	65	80	103	62	0
CIP01	75	80	90	32	0
CIP02	77	20	81	28	0

Table 5: The NSRR tests with PWR fuel

Test	Bu (MWd/KgU)	Oxide (µm)	Enthalpy (cal/g)	Pulse Width (ms)	Fail
MH1	38.9	4	47	6.8	0
MH2	38.9	4	55	5.5	0
MH3	38.9	4	67	4.5	0
GK-1	42.0	10	93	4.6	0
GK-2	42.0	10	90	4.6	0
OI-1	39.1	15	106	4.4	0
OI-2	39.2	15	108	4.4	0
HBO1	51.0	43	61	4.4	1
HBO2	50.4	35	37	6.9	0
HBO3	50.4	23	74	4.4	0
HBO4	50.4	19	50	5.4	0
HBO5	44.0	60	80	4.4	1
HBO6	49.0	33	79	4.4	0
HBO7	49.0	45	88	5.0	0

Over thousand safety tests for PWR, BWR, and MOX fuel were performed at NSRR. The current investigation uses fuel rod re-fabricated with burnups ranging from 20 to 61 MWd/kgU and the peak of enthalpy ranging from 47 to 145 cal/g at room temperature and pressure

Table 6: The NSRR test with BWR fuel

Test	Bu (MWd/kgU)	Oxide (μm)	Enthalpy (cal/g)	Pulse Width (ms)	Fail
TS1	26	6	55	6.0	0
TS2	26	6	66	5.3	0
TS3	26	6	88	4.8	0
TS4	26	6	89	4.6	0
TS5	26	6	98	4.4	0
FK1	45	16	130	4.4	0
FK2	45	19	70	6.5	0
FK3	41	24	145	4.4	0
FK4	56	22	140	4.3	0
FK5	56	22	140	7.3	0
FK6	61	25	70	4.4	1
FK7	61	25	70	4.4	1
FK8	61	25	62	7.3	0
FK9	61	25	86	5.7	0
FK10	61	25	80	5.1	0
FK12	61	25	72	5.8	0

The CABRI water loop must operate from 2000 to 2015. The project is investigating the ability of the nuclear fuel to withstand the fast power increases within a few milliseconds. The program used LWR fuels conducted from 1993 to 2002. Have several experiments carried out with UO_2 and $(Pu\text{-}U)O_2$ fuels and using pre-irradiated rods.

4. DISCUSSION AND RESULTS

The practical results in the database were used to find the peak fuel enthalpy. The failure threshold for uranium dioxide and Zircaloy-2/4 can be calculated using the correlation defined in Eq. (1).

$$H = ABu^3 - BBu^2 + CBu + D \qquad (1)$$

where: H = Enthalpy (cal/g - UO_2);
Bu = Burnup (MWd/kgU);
A = 0.00169, B = 0.1929, C = 2.86, and D = 193.4;

The results can fit the safety limit obtained by logistic regression defined in Eq. (2) Moreover, Eq. (3).

The value received is the probability of collapse. The regression is the calculation of risk if the determined value is over 0.5 considering the possible rupture of the fuel rod. The failure threshold and the probability of failure can be defined by Eq. (4), Eq. (5), and Eq. (6) for alloy M5 and by Eq. (7), Eq. (8), and Eq. (9), for alloy ZIRLO. These new correlations can be applied for irradiation levels at 40–72 GWd/MTU with cladding zirconium alloy having 1% niobium. In such cases, the pulse width must be limited to 6 ms. Figures 1 and 2 illustrate the limits.

$$q = exp(A + B.T + CBu + DOx + EW + FH + GClad) \tag{2}$$

$$p = q/(1+q) \tag{3}$$

where: p is the probability of failure;
A = -10.292; B = -0.1; C = 0.038; D = 0.46; E = -0.003; F = 0.053; G = 0.376;
T = temperature (°C);
Bu = burnup (MWd/kgU);
Ox = oxide thickness (µm);
W = pulse width (ms);
H = enthalpy (cal/g);
Clad = type of cladding: (2) for Zry-2; (4) for Zry-4; (5) for M5, and (6) for Zirlo;

$$H_{(M5)} = 316.4\exp(-0.0213\,Bu) \tag{4}$$

$$H_{(M5)} = 316.4\exp(-0.0213\,Bu) \tag{5}$$

$$p = q/(1+q) \tag{6}$$

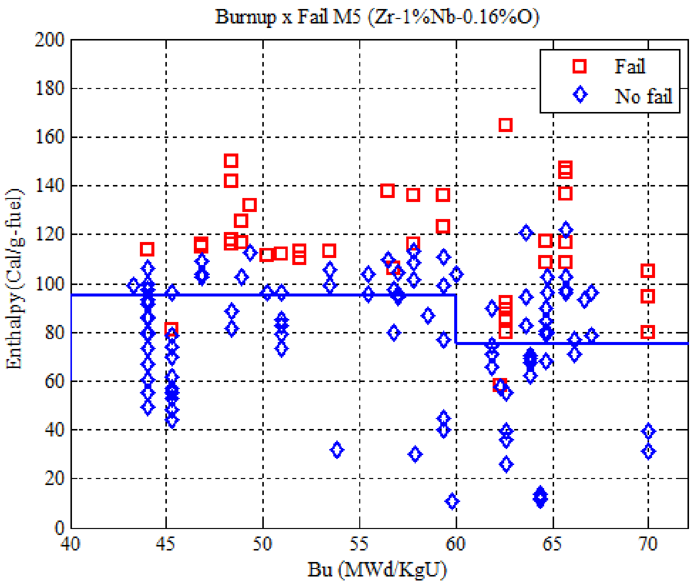

Figure 1: Safety threshold for M5 width pulse width limited at 6 ms.

Where: Bu = burnup (MWd/KgU);
A=-89.640; B=-3.822; C=0.028; D=0.008; E=-0.0043; F=0.053;
T= temperature (°C);
Ox = oxide thickness (µm);
W =pulse width (ms);
H = enthalpy (cal/g).

$$H_{(ZIRLO)} = 183.1\exp(-0.0126\,Bu) \tag{7}$$

$$q = \exp(A + BT + CBu + DOx + EW + FH) \tag{8}$$

$$p = q/(1+q) \tag{9}$$

where: T= temperature (°C);
A=13.383; B=0.1; C=0.112; D=0.316; E=0.17; F=0.149;
Bu = burn-up (MWd/KgU);
Ox = oxide thickness (µm);
W =pulse width;
H = enthalpy (cal/g).

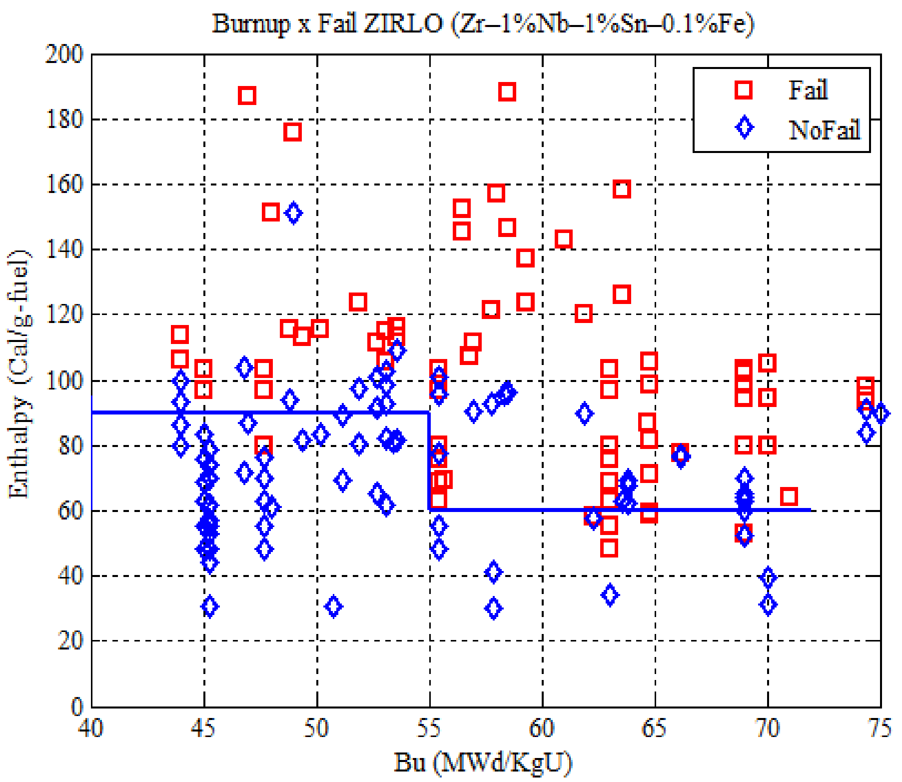

Figure 2: Safety threshold for zirlo, with pulse width limited at 6ms.

5. CONCLUSIONS

In this research, we investigated a novel approach of pattern recognition for the failure of the nuclear fuel rod under high burnup, i.e., between 45 and 72 MWd/kgU. The factor analysis reduced the dimensionality of the RIA threshold to six inputs and one output. Input variables are temperature, burnup, oxide layer thickness, pulse width, and the peak of enthalpy applied in uranium dioxide.

The regression obtained can predict accident conditions for irradiated fuel from 45 to 75 MWd/KgU if the pulse width is limited to 6 ms. In extended cycles of discharge, a decrease in the threshold can be observed. The solution represents the success of accident-tolerant fuels. New generation fuels should replace the zirconium alloys. Innovative alloys such as SiC and FeCrAl are proposed in ATF program.

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