LWR Fuel Safety Research with Particular Emphasis on RIA/LOCA and Other Conditions

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Fuel safety research at Japan Atomic Energy Research Institute (JAERI) is reviewed on the major subjects including studies on fuel behavior under postulated Reactivity Initiated Accident (RIA), postulated Loss of Coolant Accident (LOCA) and normal operating conditions. Nuclear Safety Research Reactor (NSRR) at JAERI has been utilized extensively for the studies of fuel behavior under RIA conditions. For the studies of fuel rod and cladding behavior under LOCA conditions, outpile experiments were conducted. The work on this subject has been concluded. Pellet Cladding Interaction (PCI) has been major subject on fuel integrity study during normal operating conditions. Irradiation experiments at Halden Boiling Water Reactor (HBWR) as well as code development are described.

KEYWORDS: LWR type reactors, fuel behavior, fuel integrity, fuel failure, cladding ballooning, cladding oxidation, pellet cladding interaction, reactivity initiated accident, loss of coolant, normal operating conditions, Japan

I. INTRODUCTION

Fuel safety research at JAERI has been conducted for the postulated accident conditions and for the normal operating conditions. The nuclear power plants are designed to cope with a specified range of operational states and accident conditions within the defined radiation protection requirements. Among the design basis accidents, RIA and LOCA are chosen as extreme cases.

The NSRR at JAERI has been constructed and operated for the study of fuel behavior and generation of destructive forces upon fuel failure under postulated RIA. The major experimental results and the reflection of the achievements on the safety guide in Japan are described.

Studies on the fuel behavior under postulated LOCA conditions have been conducted by extensive outpile experiments. Cladding embrittlement due to Zircaloy-steam reaction and cladding ballooning were investigated.

II. FUEL BEHAVIOR UNDER RIA

The RIA research program in Japan was initiated in 1972 as the NSRR program. Over 860 experiments have been conducted in the NSRR program to date with fresh LWR rods for various test parameters.

The NSRR is a modified TRIGA-ACPR (Annular Core Pulse Reactor) of which salient features are the large pulsing power capability, and the large (22 cm in diameter) dry irradiation space to accommodate a sizable experiment. The general arrangement of the NSRR is shown in Fig. 1. Test fuel rods contained in a test capsule were irradiated in the test cavity of the NSRR by a large pulse to simulate a power excursion of a RIA(1).
1. Major Experimental Results

(1) General Fuel Behavior

Standard test fuel rods were subjected to peak fuel enthalpies of about 30 to 450 cal/g-UO$_2$. The photographs of typical post-test fuel rods are shown in Fig. 2. In the peak fuel enthalpy range below 88 cal/g-UO$_2$, departure from nucleate boiling (DNB) and no visible change took place in post-test fuel rods. The DNB occurred at a fuel enthalpy of over 110 cal/g-UO$_2$, resulting in the oxidation of cladding surface of pelletized region. The incipient fuel failure occurred at the fuel enthalpy of about 220 cal/g-UO$_2$ due to brittle fracture of the oxidized and partially molten cladding. The cladding surface temperature reached near the melting point of Zircaloy during transient and variations of cladding
wall thickness was observed in the failed fuel rods. When the peak fuel enthalpy was over 325 cal/g-\textit{UO}_2, fuel pellets were molten and expelled into water. The fragmentation of the molten fuel caused the pressure generation and jumping of water column\(^{(2)}\).

![Fig. 2 Appearances of post-test fuel rods related with peak fuel enthalpy](image)

(2) Effects of Fuel Design Variations

The effects of fuel design parameters such as rod internal pressure, radial gap width\(^{(3)}\), gap gas composition\(^{(4)}\) etc., were examined. Among these parameters, the fuel rod internal pressure showed the most evident effect, characterized by the change of failure mode to “high temperature cladding burst”\(^{(5)}\).

(3) Effects of Cooling Environment Variations

The effects of various cooling conditions such as clustered rods geometry, system pressures \textit{etc.}, were examined. The fuel failure threshold for a rod bundle geometry and for a lower coolant subcooling was about 15% lower than that for the standard cases, a single rod under high subcooling, due to poorer heat transfer at the cladding surface\(^{(6)}\). While better cooling by applying forced convection increased the failure threshold\(^{(7)}\). Tests conducted at elevated system pressure corresponding to the operating conditions of BWRs and PWRs indicated that the failure threshold was almost the same as that observed in the atmospheric pressure capsule tests\(^{(8)}\).

(4) Waterlogged Fuel Rods

The waterlogged fuel rods generally failed at fuel enthalpy of about 100 cal/g-\textit{UO}_2 during an early stage of the power burst and causes release of the fuel pellets at very fine particles resulting in the generation of mechanical energy\(^{(9)}\).

(5) Mechanical Energy Generation

When fuel pellet is molten and ejected into the coolant through a breach of the cladding, molten \textit{UO}_2 is fragmented into small particles and causes mechanical energy generation. Sharp pressure pulses were detected at the capsule bottom upon fuel rod failure, and the ejection of water slug over the fuel region followed. Threshold enthalpy for this
mechanical energy generation is between 285 cal/g-UO₂ and 325 cal/g-UO₂. 

2. Application to Licensing Guide

"Evaluation Guide for Reactivity Initiated Events in Light Water Power Reactors" was issued by the Nuclear Safety Commission in January 1984, based on the knowledge obtained in the NSRR experiments. The acceptable fuel design limit has been determined primarily based on the single rod test data with non-pressurized and pre-pressurized fuels shown in Fig. 3. For an accident, the guideline requires that the maximum radial average fuel enthalpy shall not exceed the value of 230 cal/g-UO₂ to prevent the generation of mechanical energy. In addition, the guideline specifies the requirement for the evaluation of the influence of the rupture of waterlogged fuel rods which are assumed to exist in the core during normal operations.

![Fig. 3 Fuel rod failure threshold in NSRR experiments and licensing limit for reactor transient](image)

3. Future Plan

The NSRR experiments conducted so far are with fresh fuel rods. The experiments conducted with pre-irradiated rods are only 10 cases in SPERT and PBF experiments conducted in the United States. So, the information on the effects of the burnup is too limited to define the thresholds for fuel failure and for fuel fragmentation. As a second phase of the NSRR experiments, the experiments with pre-irradiated fuel rods is being planned. Modification of the facilities is underway, and the experiments will be started early 1989.

III. STUDIES ON FUEL CLADDING BEHAVIOR UNDER LOCA

The acceptance criteria for capability of an emergency core cooling system during LWR LOCA require that coolable geometry of the fuel assemblies must be kept in the course of the accident. In order to determine whether or not coolable geometry is kept, following experiments have been performed:

1. Zircaloy-steam Reaction and Ductility of Reacted Zircaloy Tubing

The isothermal reaction rates of Zircaloy-steam were measured at a few laboratories in Japan. All results are lower than Baker-Just's reaction rate. JAERI developed a computer code, PRECIP-2 which could calculate the Zircaloy-steam reaction, and it was verified by experiments under transient temperature.
JAERI found that Zircaloy cladding picked up considerable amount of hydrogen under LOCA conditions, and was severely embrittled. This mechanism was studied by the following experiments\(^{17}\); (1) reaction of Zircaloy tubing with stagnant steam, (2) reaction of Zircaloy tubing with flowing hydrogen/steam mixed gas, and (3) burst/oxidation of simulated fuel rods in flowing steam. A typical embrittlement behavior is shown in Fig. 4.

From the experiments, the mechanism of abnormal inner surface oxidation (hydrogen uptake) was clarified as follow:

After the cladding burst, inner surface of the cladding is oxidized by penetrating steam and generated hydrogen at a location somewhat apart from the ruptured position remains in the atmosphere. Hydrogen content in the atmosphere increase with time. When the hydrogen content reaches to a critical value, hydrogen absorption starts.

Based on the above results, it seemed to be necessary to review the then existed embrittlement criteria for the Zircaloy cladding because this phenomenon was not taken into account in the criteria. Thus an additional experiment was conducted based on the assumption that if claddings which absorbed hydrogen did not fail during the reflooding period under an axially restrained condition, coolable geometry of a fuel assembly could be kept.

A simulated fuel rod was burst and subsequently heated in flowing steam. After maintaining the rod during specified time, it was quenched by water under obstruction of

*Fig. 4* Variations of oxide layer thickness at inner surface, ductility and absorbed hydrogen content of Zircaloy fuel cladding ruptured and oxidized at 1,332 K for 240 s in flowing steam with flow rate of 267 g/m²s as a function of distance from a rupture opening.
2. Single Rod and Multirods Burst Tests

Under a LOCA, ballooning of fuel claddings decreases coolant flow area in a channel, and it may influence the reflooding speed of emergency core cooling water. In order to estimate the upper limit of channel restriction, burst tests of single rods and multirods were performed.

Direct and indirect heating methods were applied in single rods tests. In some of the tests, a specimen was surrounded by 8 heaters. A specimen of multirods was consisted of 7x7 fuel assembly, and the heating was indirect. By the tests, following results were obtained:

1. The maximum deformation was observed when the burst occurred around $\alpha$ and $\alpha+\beta$ boundary of Zircaloy.

2. Large azimuthal temperature gradient was produced in the claddings adjacent to the control rods. However, this phenomenon did not necessarily decrease the amount of ballooning. It seems that the temperature difference between deforming rod and the surrounding rods dominantly influenced on the amount of ballooning.

3. The upper limit of coolant flow area restriction in a bundle under LOCA was estimated as about 80%.

A computer code, FRETA-B which could calculate temperature and deformation of fuel claddings in a bundle during LOCA was developed and verified by experimental results obtained in JAERI and foreign countries.

IV. STUDIES ON FUEL BEHAVIOR UNDER NORMAL OPERATING CONDITIONS

The JAERI has conducted irradiation experiments at HBWR in Norway and at Japan Material Test Reactor (JMTR) of JAERI. The major effort has been given to the study of PCI including inpile fuel rod deformation measurements at HBWR. The recent JAERI's experiments at HBWR on this subject are described in this chapter. The experimental conditions cover steady state irradiation, power ramping and power cycling. The data obtained from the experiments together with other sources were used for development and verification of the computer code FEMAXI of JAERI.

1. Experimental Procedures

Fuel rod behavior during power ramping and cycling has been investigated in a power ramping rig with measurements of rod diameter, elongation, eddy current detection of the cladding cracks and internal pressure under HBWR, simulated BWR or PWR coolant conditions at HBWR. Power rampings were conducted with stepwise power increase. Power cycling experiments were conducted in the power ramping rig with attachment of a special cyclic control systems.

Base irradiation to accumulate burnup of the rods was conducted using instrumented
base irradiation rigs of HBWR, BWR or PWR conditions.

2. Major Experimental Results(25)~(34)

(1) Influence of Design and Fabrication Parameters

The influence of pellet shape, pellet and cladding gap size, cladding thickness etc., were investigated. Regarding on influence of pellet shape, the ridges formed on the cladding due to Pellet Clad Mechanical Interaction (PCMI) are larger and more regular with dished pellet than with flat pellets. Regarding the influence of the gap size, it was found that PCMI begins at lower rod power and that the extent of PCMI at high power is larger in small gap rods. Regarding the influence of cladding thickness, a rod with thin cladding develops larger ridges than the one with thicker cladding, is mainly attributed to the effect of creep-down of the cladding.

(2) Influence of Burnup

The influence of burnup on PCMI results mainly from pellet-cladding gap closure and from the increase of the elastic limit of the cladding. This means that the change in rod diameter and length at peak power during ramping are generally large after base irradiation.

(3) Influence of Power Cycling

Regarding the deformation behavior, the cycled and non-cycled rod showed similar diametral profiles for the test period. The fission gas release (FGR) of PWR rod as measured by puncturing during post irradiation examination are very similar for the cycled (620 cycles) and non-cycled rods, i.e., 18.9 and 18.1%, respectively. In more detail inpile pressure gauges seemed to show that FGR of the cycled rod occured at every cycle, while in the non-cycled rod FGR mainly occured during power decrease.

(4) PCI Failure

PCI failures were observed in some rods. Local deformation of the cladding of failed rods was found to be small until the crack penetrated the wall. A failure map, expressed as maximum linear heat rating vs. burnup is obtained and compared with the data from Halden Reactor Project experiments and from other international projects with good agreement.

3. FEMAXI Fuel Analysis Code Development

The FEMAXI code applies axisymmetric finite element method for the analysis of local deformation and pellet cladding contact problem in detail. FEMAXI-IV is the latest version and can handle both steady state and transient operating conditions.

FEMAXI-III and IV have been verified by the data from JAERI’s power ramping experiments at HBWR as described above and by the recent data from Halden Reactor Project and other international projects. The code calculation were found useful for understanding the fuel behavior as well as evaluation of fuel integrity. A typical example of application of the code calculation is given as follows.

Clear ridge formation was observed on PWR rods during ramping as shown in Fig. 6. The solid line represents the measured profile and the dotted line does the calculated one. When the power is increased, secondary ridge formation became clear by calculations.
formation was observed as seen in the lower figure. The stress and strain from the calculation were examined in detail and it was found that creep of UO₂ occurred towards dished space at high power. Accordingly, stress near the pellet interfaces relaxed while the stress accumulated at the mid pellet position due to thermal expansion. This resulted in the apparent secondary ridge formation.

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