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EVALUATION OF LOCAL ACCIDENTS IN LMFBR

Yoshihiro KIKUCHI, Hiroshi MIZUTA, Hidekazu YOSHIKAWA,
Masao HORI and Kenji TANAKA

Power Reactor and Nuclear Fuel Development Corporation
Japan

ABSTRACT

This paper deals with the evaluation of local faults occurred in a fuel subassembly and a steam generator of a liquid metal cooled fast breeder reactor (LMFBR) system.

Fuel pin integrity under various abnormal conditions is being confirmed by both out-of-pile and in-pile tests. Out-of-pile experiments were conducted to evaluate the mechanical and thermal effects of fission gas ejection on adjacent pins and a wrapper tube. Out-of-pile experiments were conducted to measure local temperature rises due to fuel pin contact and local flow blockages, and the measured temperature rises agreed fairly with the predictions by computer codes. Extensive studies were conducted on local and bulk sodium boiling phenomena and the experimental results were in satisfactory agreement with the computer code predictions. Out-of-pile experiments are being conducted to measure the magnitude of pressure pulses generated in molten fuel-coolant interactions. The integrity of wrapper tubes to high pressure was examined by gunpowder explosive tests in water and the possibility of meltthrough to adjacent subassemblies were evaluated analytically. Various local anomaly detection methods are being studied along with the above experiments.

Tests and analyses of sodium-water reaction also have been conducted to evaluate the integrity of a steam generator.

1. INTRODUCTION

Evaluation of various local faults is important for safety considerations in the liquid metal cooled fast breeder reactor (LMFBR) system, since a local initiating fault may successively cause the failure propagation and finally lead to a serious damage of the total system. Extensive studies have been carried out in this connection for analyzing the probable accident sequence and consequence caused by local faults as well as developing the local anomaly detection methods.

This paper summarizes the evaluation techniques for the local faults in a fuel subassembly of the fast reactor core and the sodium-water reaction in a steam generator, with an emphasis on the former.

Critical areas in the local core faults studied thus far are : fuel pin integrity, fission gas ejection, fuel pin contact, local flow blockages, sodium boiling, molten fuel-coolant interaction, wrapper tube integrity and local anomaly detection methods.

In addition, the systematic studies of sodium-water reaction in a steam generator will be informed.

2. LOCAL CORE FAULTS

2.1. Fuel pin integrity

Fuel pin integrity under various abnormal conditions is the first barrier against local core faults. Electrically heated out-of-pile tests were performed to examine the cladding integrity under loss-of-flow conditions, simulating various burnup effects to the cladding by equivalent physical and mechanical treatments. It was revealed that the cladding integrity can be maintained up to the tempereture of 1000°C.

In addition to the out-of-pile tests mentioned, in-pile tests by using a Nak capsule in GETR* are proceeding for confirming MONJU** fuel pin integrity under abnormal loss-of-flow and local flow blockage conditions. Total 15 tests are being conducted on fresh and preirradiated fuel pins.

2.2. Fuel pin contact

Experimental studies were conducted of the heat transfer in sodium flowing in an electrically heated seven-pin bundle in which a pin was bowed to form a point contact[1]. The measured temperature rise at the contact point is shown in Fig.1 along with the analytical fitting by the PICO code. If the experimental results are extrapolated to actual reactor conditions, the temperature rise due to a point contact is estimated to be less than 40°C.

The experiments have recently been conducted in a line contact condition. The measured temperature rise due to a line contact is higher than that due to the previous point contact and then is estimated to be less than 60°C under reactor conditions.

2.3. Local flow blockage

Sodium loop experiments were conducted of local temperature rise due to a non-heat-generating blocakge in an electrically heated seven-pin bundle[2]. The central six-subchannel blockage (blockage fraction of 42 %) brings the hottest point immediately downstream from the blockage as shown in Fig.2, which also shows the calculation by the LOCK code. If the experimental results are extrapolated to the reactor conditions, the temperature rise due to the six-subchannel blockage is estimated to be less than 250°C.

* General Electric Testing Reactor
** Japanese Prototype Fast Breeder Reactor

Since a larger-scale flow blockage, even if it is less probable, may cause sodium local boiling or further bulk boiling, recent efforts are directed toward the larger-scale experiments in 19-and 37-pin bundles. In parallel to the sodium loop experiments water loop experiments in a 61-pin expanded mockup are also in progress for investigating the detailed flow distribution behind the blockage, which serves the development of a multi-dimensional thermohydraulic analysis code.

All the experiments mentioned above have been conducted using grid-type pin bundles. Local blockage effects in the wire wrapped subassemblies also are being investigated by the experiments and analyses, though the formation of a large planer blockage is much less likely than the grid-type subassemblies. The wire wrapped subassemblies will be used in the first core of MONJU.

2.4. Fission gas ejection

The fission gas ejection tests were first carried out in a subassembly using a water loop in order to evaluate the mechanical damage to the adjacent pins and the wrapper tube wall. The observed mechanical effects were rather insignificant[3].

The second series of experiments were conducted to evaluate the thermal effects due to gas injection into sodium flowing in a 37-pin bundle, which consisted of a central gas-injector pin, seven electrically heated pins and other dummy pins[4]. The rapid temperature rise due to gas blanketing observed on the heated pin surfaces at the axial height of gas injection is shown in Fig.3, along with the fitting by a transient heat transfer model. Extrapolation of the experimental results to actual reactor conditions reveals the temperature rise due to the gas blanketing is small and well within tolerable limits.

Further experiments in locally blocked conditions are now being conducted.

2.5. Sodium boiling

Various experimental data concerning sodium boiling have been extensively accumulated through electrically heated single- and seven-pin experiments under steady-state, loss-of-flow and locally blocked conditions[2,5,6,7]. Some examples are shown for incipient-boiling superheat in Fig.4 and residual film thickness in Fig.5. The observed two-phase flow pattern in the steady-state boiling experiments under forced convection was in the sequence of bubbly flow, slug flow and annular (mist) flow in sodium as in the case of water. In the loss-of-flow experiments, however, the single-bubble slug-ejection pattern was dominant.

Besides the experiments analytical codes are developed for the evaluation of sodium boiling. Their modeling and coding statuses are shown in Table I. Comparison between experiments and analyses by the NAIS-P2 code is shown in Fig.6.

Larger-scale experiments both in 19- and 37-pin bundles started in 1976, with the purpose of obtaining the detailed knowledge about the transition codition of local boiling to bulk boiling, because the previous seven-pin experiments lack a prototypical radial temperature distribution.

2.6. Molten fuel-coolant interaction

Out-of-pile experiments for the UO_2-Na system have been conducted in order to understand the fundamental phenomenology of molten fuel-coolant interaction (FCI) processes. The first series, molten fuel dropping experiments into liquid sodium pool, resulted in the following : (1) excessive fragmentation but rather mild pressure generation (<3.75 bars), (2) a log-normal particle size distribution around the mean diameter of about ten microns[11].

In order to allow a more intimate contact between molten UO_2 and sodium, the second series of Joule heated experiments are being carried out with short pins being heated directly in a sodium capsule and loopsule [12]. Although the experimental data are insufficient to make final conclusions, the fuel motion and the FCI phenomena are rather affected by geometrical constraints, judging from the results of closed and open type capsule tests.

Our present knowledge of FCI taken into account, local FCI effects within a subassembly have been evaluated by combinating the conservative Cho-Wright model for FCI with the DETECT code[13] for evaluating subassembly thermohydraulics.

2.7. Subassembly-to-subassembly failure propagation

The integrity of the subassembly wrapper tube wall to the pressure pulse was examined by gunpowder explosion tests for single and seven wrapper tubes in water[14]. The dynamic deformation of the wrapper tube wall was also evaluated by the HEXATUBE code. It was concluded that the subassembly wrapper tube wall has a sufficient strength against any strong pressure pulse caused by the local accidents.

Concerning the possibility of thermal failure propagation, a preliminary analysis by the SARUP-3F code indicates that the meltthrough of the incident subassembly wall and the propagation to the adjacent subassemblies are slow enough for the suitable countermeasure to be taken [15].

2.8. Anomaly detection methods

A second safety strategy is followed by developing an appropriate in-core instrumentation which enables detection of local anomalies early enough to prevent any further core damage. Thermocouples, flowmeters and acoustic instruments are under development for further application to MONJU.

2.8.1 Temperature fluctuation

The quantitative evaluation of the flow decrease caused by a local flow blockage in water loop experiments revealed only 5 % decrease in flow for a 44 % central blockage in flow area. This indicates that small local blockage cannot be detected by the normal subassembly outlet instrumentation (thermocouple/flowmeter).

Concerning the temperature fluctuation as an altenative candidate, comparison of the outlet temperature fluctuations obtained from the blocked and unblocked seven-pin experiments led to the fact that the blockage formation would considerably intensify the low-frequency spectrum (<10Hz), and moreover, increase the measured RMS by a factor of two to three[16], as shown in Fig.7. This fact encourages detectability.

2.8.2 Acoustic noise[2,6]

If one examines the acoustic noise data with sodium boiling in Fig.8, the acoustic intensity begins to increase when the sodium flow enters into the two-phase regime. The noise intensity with sodium boiling is much higher than that with water boiling.

The peak observed at the low-frequency hertz ranges was due to the repetition of bubble formation and collapse. In the high-frequency kilohertz ranges, however, resonance peaks were superposed on a smooth curve with a broad peak at approximately 7 kHz.

These results indicate that the measurement of acoustic noise signals associated with sodium boiling phenomena is promising for early detection of local sodium boiling in LMFBR fuel subassemblies.

2.8.3 Flow fluctuation[2]

Examination of the RMS value of outlet flow oscillation in the locally-blocked bundle experiment also indicates the possibility of local boiling detection.

3. SODIUM-WATER REACTION IN STEAM GENERATOR

3.1 Design basis leak

In the design and safety evaluation of the MONJU plant, it is assumed that the Design Basis Leak, the maximum sodium-water reaction accident, is the one caused by the water leak from the double-ended ruptures of four tubes, where the initial failure is assumed to be less than one tube rupture and the secondary failure to be less than three tubes ruptures. Even in the maximum accident, the boundary wall between the primary and secondary circuits in the intermediate heat exchanger should withstand the pressure propagated from the steam generator, and also the integrity of the steam generator shell and the secondary circuit components should be kept.

3.2. Evaluated results

For the safety evaluation of the MONJU steam generators, the following items are being demonstrated and confirmed by the sodium-water reaction tests and analyses:

(1) The confirmation of the maximum scale of the accident[18,20,21]

It is demonstrated that the maximum accident caused by the initial failure of less than one tube rupture is less than four tube ruptures. For the above purpose, tests and analyses have been conducted on the following itmes.

(i) The time of the failure propagation and the extent of the secondary failure due to the initial failure of less than one tube rupture.

(ii) The time of the leak detection and termination for the initial failure of less than one tube rupture.

(2) The confirmation of the effect of the maximum accident[17,19,21,22]

It is demonstrated that the effect of the maximum accident (four tube ruptures) on the steam generators, the intermediate heat exchanger and other secondary circuit components is within the tolerable limit. For the above purpose, tests and analyses have been conducted on the following items.

(i) The sodium-water reaction pressure and temperature in the steam generator for the leak rate corresponding to the one-plus-three tube ruptures. Specifically, the initial pressure spike, strain of the

steam generator shell, quasi-static pressure, performance of pressure relief system, maximum temperature of the steam generator shell etc.
(ii) The propagation of pressure to the secondary circuit and the intermediate heat exchanger for the leak rate corresponding to the one-plus-three tube ruptures.

From the tests and analyses described above, it is shown that the present type of MONJU steam generator can be designed to withstand the pressure, temperature and other effects of the sodium-water reaction accidents.

4. CONCLUSIONS

Overall evaluations concerning the local core faults mentioned in chapter 2 cover all the areas necessary for the fast reactor safety within our present state of knowledge, and they generally confirm the safety margin for various local faults.

Evaluation of sodium-water reaction, which is described in chapter 3, shows that the integrity of steam generator and other secondary components can be kept in any conceivable accident.

Future works for the problems left in these areas are summarized as follows :

(1) Verification of the fuel pin integrity by the systemmatic in-pile test programme. This task would have a high importance from the viewpoint of effective LMFBR system operation.
(2) More sophistic evaluation of the severer and hence less probable local accident sequence from intra-subassembly failure to trans-subassembly failure.
(3) Effective synthesis of the local anomaly diagnostics which will detect local faults reliably and early in time and which hopefully will yield the information needed for counteracting these anomalies.
(4) Development of a leak detection system with more reliability and faster response, so as to make the damage to steam generator minimum.

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TABLE I Sodium Boiling Analysis Codes

Code Name	Modeling Status	Coding Status
NAIS-P2[5]	Single pin Single-bubble slug (uniform vapor)	Good Agreement with experiment
CRUSH[8]	Single pin Multi-bubble slug (uniform vapor	Good agreement with experiment
SPACE[9]	Multi pins Multi-bubble slug (uniform vapor)	Version-up phase to local boiling analysis
PAPAS[10]	Single pin Multi-bubble slug (pressure-gradient vapor)	Good agreement with experiment.Systemized as a module of LMFBR safety analysis code

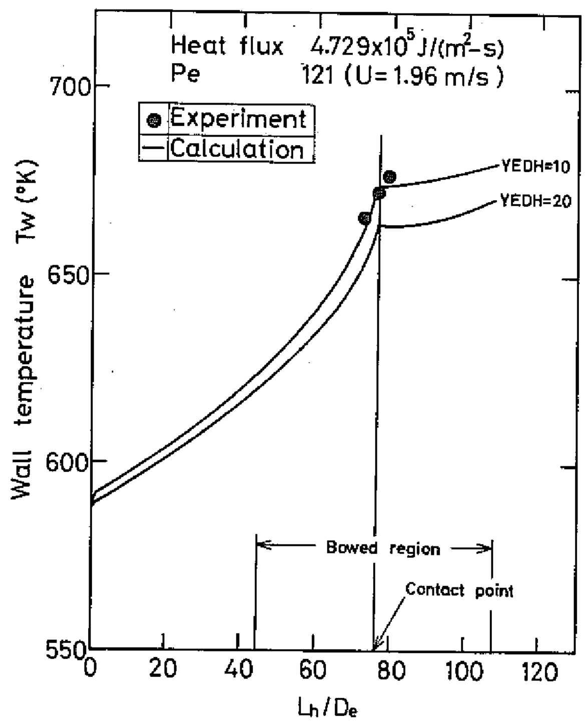

Fig. 1 Comparison of longitudinal temperature distribution at contact point observed in experiment with PICO code calculation

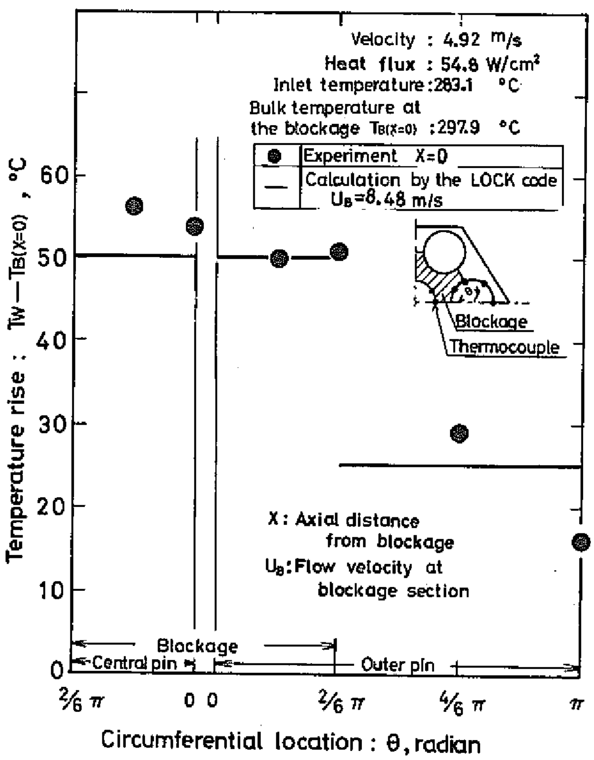

Fig. 2 Comparison of circumferential temperature distribution at blockage section observed in experiment with LOCK code calculation

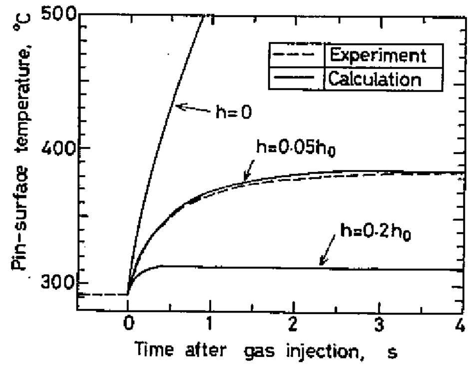

Fig. 3 Comparison of temperature rise due to gas blanketing observed in experiment with SURFACE code calculation

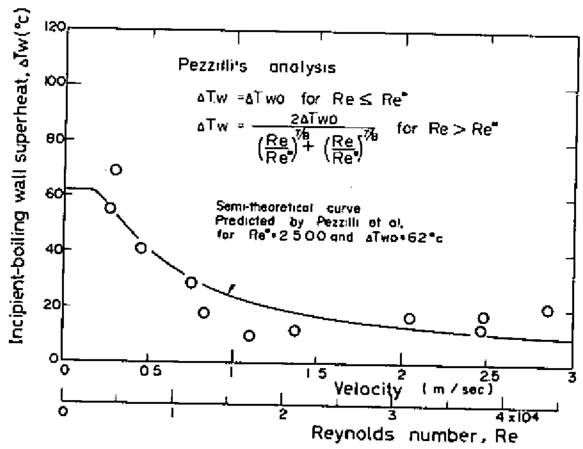

Fig. 4 Effect of flow velocity on IB wall superheat

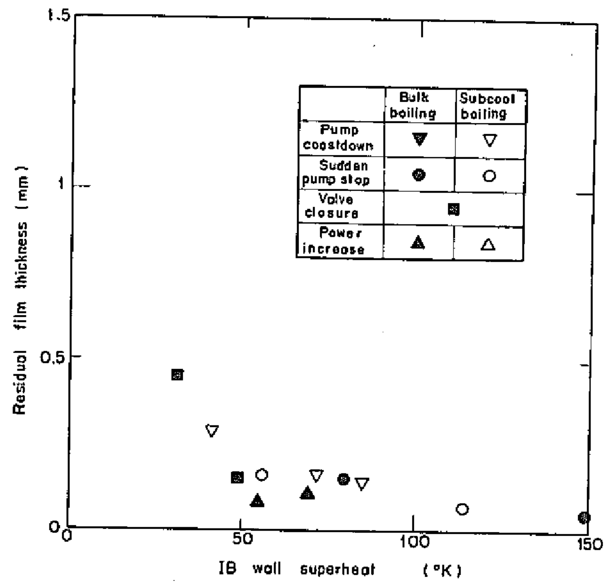

Fig. 5 Effect of IB wall superheat on initial residual liquid film thickness

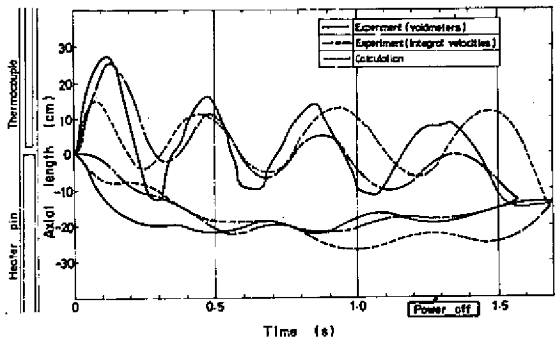

Fig. 6 Comparison of boiling pattern observed in single-pin experiment with NAIS-P2 code calculation

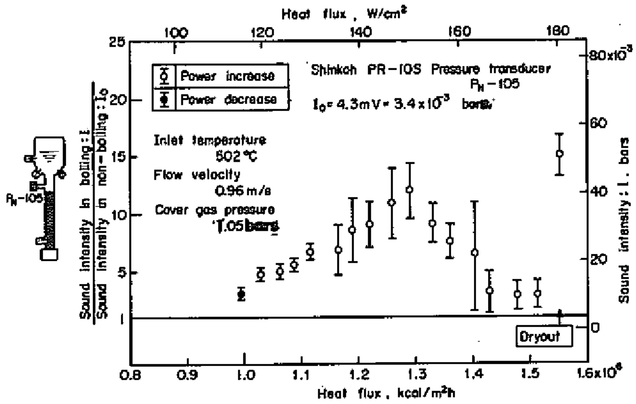

Fig. 8 Effect of heat flux on intensity of boiling acoustic noise in locally blocked seven-pin bundle

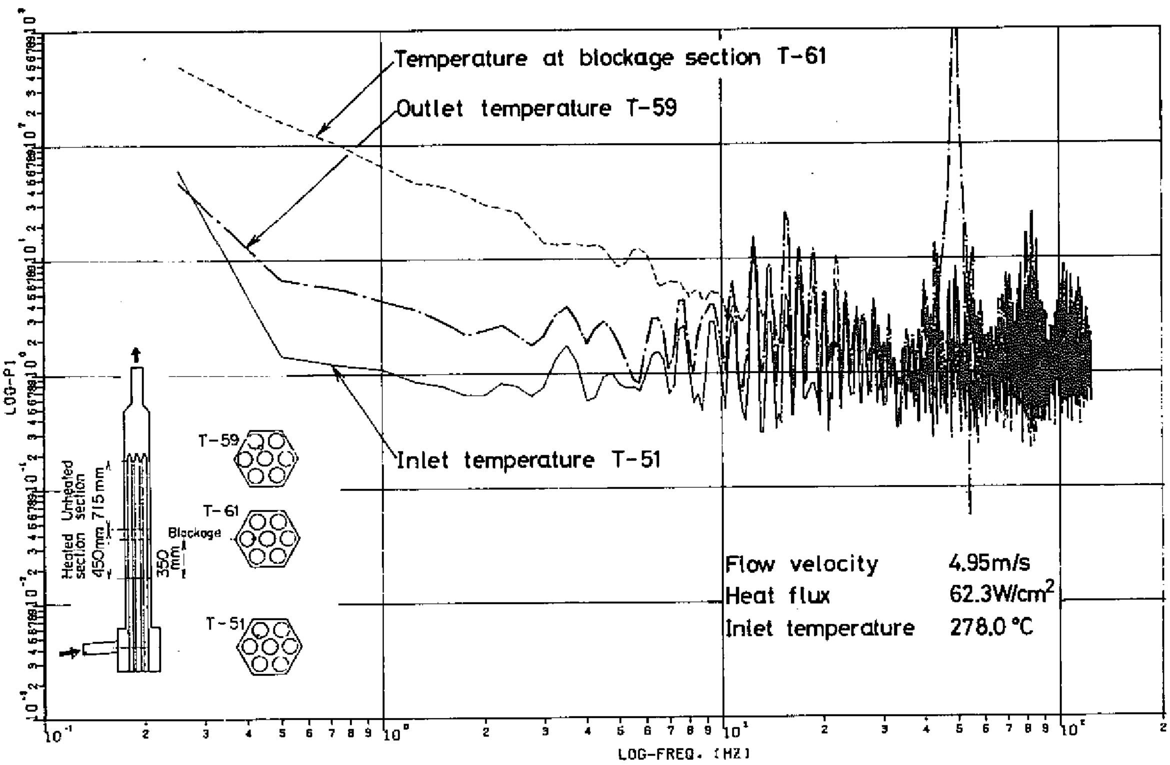

Fig. 7 Auto power spectra of temperature fluctuations in locally blocked seven-pin bundle