Report of the 17th International Workshop on Nuclear Safety and Simulation Technology (IWNSST17)

YOSHIKAWA Hidekazu[1, 2]

1. Symbio Community Forum, Co. Research Institute of for Applied Sciences, 49 Tanaka-Ohi-Cho, Sakyo District, Kyoto 606-8202, Japan (yosikawa@kib.biglobe.ne.jp)
2. College of Nuclear Science and Technology, Harbin Engineering University, 145-1 Nantong Street, Nangang District, Harbin 150001, China

Abstract: The 17th International Workshop on Nuclear Safety & Simulation Technology (IWNSST17) was held in January 21, 2014 at Kyoto University, in Kyoto, Japan. This one-day workshop was motivated to exploit advanced safety researches for nuclear power plant (NPP) , by a unique synergetic collaboration of basically two different disciplines: material science and systems sciences. There were ten invited presentations at the ISSNP2013, and the subject of the presentations ranges from (i)material corrosion issue of NPP components, (ii)application of augmented reality technology for NPP decommission,(iii)functional modeling method for plant control system, (iv) intrinsic understanding of Fukushima Daiichi accident phenomena based on simple physical model, (v) system reliability evaluation method for PWR safety system, (vi) automatic control system design for small modular reactor, and (vii) validation of computerized human-machine interface and digital I&C for PWR plant. This article provides the overview of the IWNSST17 with giving condensed summaries of all invited presentations given by international experts.

Keyword: material corrosion; augmented reality; Fukushima accident; functional modeling; reliability evaluation; digital I&C systems; human-machine interface

1 Introduction

The 17[th] International Workshop on Nuclear Safety and Simulation Technology (IWNSST17) was held on January 21, 2014 at Conference room 3, Clock Tower Centennial Hall, Yoshida campus, Kyoto University, Kyoto, Japan, by taking the chance of several international visitors to our Symbio Community Forum in Kyoto, Japan, as a new trial to exploit advanced safety research areas for nuclear power plant (NPP) , by a unique synergetic collaboration of basically two different disciplines: material science and systems sciences. The purpose of this article is to give readers of this journal (IJNS) a comprehensive summary of this unique workshop.

2 Overview of workshop program

The 17[th] International Workshop on Nuclear Safety and Simulation Technology (IWNSST17) was organized by Symbio Community Forum, Kyoto, Japan, with the collaboration of College of Nuclear Science and Technology of Harbin Engineering University, China. The workshop program of IIWNSST17 is as shown in Table 1. The list of the workshop organizers as well as the ten invited speakers from four countries (U.S.A., China, Denmark, and Japan) is given in Table 2. The workshop was closed to the pre-registered members. So total number of participants was 20. Photo 1 shows a scene of the workshop room, with the group photo of all participants at ISSNP2013 by Photo 2.

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Table 1 Time table of IWNSST17 workshop program

Time	Items	Speakers
9:30	Opening address	
Morning Session 9:40~12:20 Chair: Prof. Morten Lind		
9:40	"Corrosion and Oxide Layer Growth Modeling Using Deterministic and Stochastic Methods"	Prof. Chen Yitung
10:20	"Reliability Assessment for Detecting and Sizing Pipe Wall Thinning and its Application to Risk Management"	Prof. Fumio Kojima
11:00	" Corrosion of Structural Materials and Electrochemistry in High Temperature Water of Nuclear Power Systems"	Dr. Shunsuke Uchida
11:40	" Introduction of AR Research Activities for Decommissioning of NPP"	Prof. Hiroshi Shimoda
12:00	Lunch break	
Afternoon session (I) 13:00~14:40 Chair: Prof. Takeshi Matsuoka		
13:00	"Short Introduction of Automation Researches at DTU"	Prof. Ole Ravn
13:40	" Functional Modeling of Control Systems"	Prof. Morten Lind
14:20	Coffee break	
Afternoon session (II) 14:40~17:20 Chair: Mr. Takashi Nitta (JAPC)		
14:40	" Post-facta analysis of Fukushima Daiichi Accident by Simple Physical Model"	Dr. Fumiya Tanabe

15:20	"Reliability analyses of PWR safety systems by the GO-FLOW methodology"	Prof. Takeshi Matsuoka
16:00	" The Automatic Control Design and Simulation of Reactor Control System in Small Modular Reactor"	Mr. Longtao Liao
16:40	" Development of Mitsubishi Computerized Human Machine Interface and Digital I&C system for PWR Plants"	Mr. Koji Ito
17:20	Closing remarks	

Table 2 List of workshop organizers and invited speakers

Name	Affiliation	Country
Prof. Hidekazu Yoshikawa	/Symbio Community Forum/Harbin Engineering University (HEU) (Workshop organizer)	Japan China
Mr. Takashi Nitta	Japan Atomic Power Company (JAPC) Symbio Community Forum (Workshop organizer)	Japan
Mr. Tamiya Yoshida	Symbio Community Forum (Workshop organizer)	Japan
Prof. Yitung Cheng	University of Nevada Las Vegas	U.S.A.
Prof. Fumio Kojima	Kobe University	Japan
Dr. Shunsuke Uchida	Institute of Applied Energy	Japan
Prof. Hiroshi Shimoda	Kyoto University	Japan
Prof. Ole Ravn	Danish Technical University (DTU)	Denmark
Prof. Morten Lind	Danish Technical University (DTU)	Denmark
Dr. Fumiya Tanabe	Research Institute of Sociotechnical Safety (SOCTEX)	Japan
Prof. Takeshi Matsuoka	Utsunomiya University	Japan
Mr. Longtao Liao	Nuclear Power Institute of China (NPIC)	China
Mr. Koji Ito	Mitsubishi Heavy Industries, Co.Ltd (MHI)	Japan

Photo 1 A scene of the workshop room

3 Record of morning session

There were four papers presented in the morning session. The first three papers were all related with material science, while the last paper was related with the application of augmented reality technology for the decommissioning of nuclear power plant. The summaries of the four presentations are given in this chapter.

Photo 2 Group photo of all participants

3.1 Corrosion and oxide layer growth modeling using deterministic and stochastic methods

Prof. Chen Yitung (University of Nevada Las Vegas) presented his research on corrosion and oxide layer growth modeling mainly concerned with use of lead-bismuth eutectic for next generation reactor. The summary of his presentation is given in the following paragraphs.

He initiated his presentation by introducing six types of Next Generation Reactors: They are:

(i)Very-High Temperature Reactor (VHTR):

a graphite-moderated, helium-cooled reactor with a once-through uranium fuel cycle,

(ii)Supercritical-Water-Cooled Reactor (SCWR):

a high-temperature, high-pressure water-cooled reactor that operates above the thermodynamic critical point of water,

(iii)Gas-Cooled Fast Reactor (GFR):

it features a fast-neutron-spectrum, helium-cooled reactor and closed fuel cycle,

(iv)Lead-Cooled Fast Reactor (LFR):

it features a fast-spectrum lead of lead/bismuth eutectic (LBE) liquid metal-cooled reactor and a closed fuel cycle for efficient conversion of fertile uranium and management of actinides

(v)Sodium-Cooled Fast Reactor (SFR):

it features a fast-spectrum, sodium-cooled reactor and closed fuel cycle for efficient management of actinides and conversion of fertile uranium, and

(v)Molten Salt Reactor (MSR):

it produces fission power in a circulating molten salt fuel mixture with an epithermal-spectrum reactor and a full actinide recycle fuel cycle.

Among those next generation reactor concept, the subject of his presentation is related with realization of LFR.

The National Materials Crosscut Program (NMCP) in the U.S. expects the candidate materials for the nuclear energy applications meet the following design objectives: (1) acceptable dimensional stability including void swelling, thermal creep, irradiation creep, stress relaxation, and growth; (2) acceptable strength, ductility, and toughness; (3) acceptable resistance to creep rupture, fatigue cracking, creep-fatigue interactions, and helium embrittlement; and (4) acceptable chemical compatibility and corrosion resistance (including stress corrosion cracking and irradiation-assisted stress corrosion cracking) in the presence of coolants and process fluids.

The fundamentals of corrosion mechanism and oxide layer growth need to be carefully studied in order to understand how it happens and how we can protect the engineering systems and extend the material life span. It is especially important in the application of nuclear energy because corrosion could lead to the loss of coolant accidents (LOCAs) and the reactor damages. The research efforts in studying the corrosion phenomena have been used the different microscopic, meso-scopic, and atomic levels to understand the corrosion mechanism of how and why it occurs under the different fluid flow conditions. The deterministic method has been firstly sought and studied. The simplified analytic solutions have been derived and compared to the numerical and experimental results. The stochastic method has also been used and developed in order to simulate the oxide layer growth.

The advantages and disadvantages of using deterministic and stochastic methods were discussed in his presentation. Then, the successful application of the developed models was introduced for the lead-bismuth eutectic system and the supercritical water system for the different materials and super alloys, with the reasonable results and good agreements compared with available experimental measurements

Multi-scale method including deterministic and stochastic approaches are currently under developing in his laboratory to study fluid mechanics coupled with erosion, corrosion, and oxide layer growth to be able to predict the material life span under the flow conditions. The final goal of the numerical approaches is to accomplish and satisfy the National Materials Crosscut Program (NMCP) in the U.S.

Photo 3 Prof. Yitung Cheng

3.2 Reliability assessment for detecting and sizing pipe wall thinning and its application to risk management

Prof. Fumio Kojima (Kobe University) presented his lecture with the order of (i)Pipe wall thinning management, (ii) E-MAT based NDE system, (iii) Probability of detection (POD), and (iv) Application to risk management. The summary of his presentation is given in the following paragraphs.

Pipe wall inspection is aimed at providing a life management process ensuring replacement or repair prior to in-service failure. The application of condition monitoring (CM) to pipe wall inspection plays essential roles in developing existence instrumentation of measurement equipment and together with better performance for optimizing maintenance procedures of piping system.

In his research, CM is applied to pipe wall thinning monitoring with hybrid use of ultrasonic guided wave (GW) and electromagnetic acoustic transducer (EMAT). GW enables long-range inspection and is an efficient technique for screening position and direction of any defect on pipe. EMAT has advantages on continuous surveillance on the pipe wall thickness. Both techniques allow the remote capabilities of inspection performance and, as a result, the proposed inspection technique is feasible for condition based monitoring of pipe wall inspection.

The main part of the research is concerned with the reliability assessment of the proposed inspection technique using probability of detection (POD). In GW, the inspection process involves randomness due to variability in inspection conditions, including inspection strategies for test signals, difficulties associated with the location, and size of pipe wall pinning. Taking into account for these factors, a simple inspection model was developed with the appropriate threshold values that are obtained by maximizing the matching detection events between the model and the inspection data. In GW test, the threshold value of detecting model can be given by POD curve based on hit/miss analysis. In EMAT, a reliability assessment method was discussed for the pipe wall thinning measurements using EMAT-EMAR. The POD function was then evaluated within the common versus an approach.

The final part of his lecture was devoted to applicability and the validity of the POD based assessment method. The structural integrity and safety margins were reported to be maintained for the piping systems by providing the acceptance criteria for wall thinning. Such safety margin can be derived by means of the hybrid use of predicting and monitoring. A predictive model for pipe wall thinning rate is formulated with the uncertainty qualification. The prescribed model reveals quite complicated manner that depends on the piping system geometry, operational records in service, environmental conditions, *etc*. The verification and validation tests are usual procedures in the proper modeling issues. Consequently, the safety margin can be evaluated by following the logarithmic normal distribution.

3.3 Corrosion of Structural Materials and Electrochemistry in High Temperature Water of Nuclear Power Systems

Dr. Shunsuke Uchida (Institute of Applied Energy) presented his lecture on the subjects of (i)Optimal water chemistry,(ii)Theoretical approaches towards quantifying interaction of materials and water, (iii)Electrochemistry, (iv)Electrochemical corrosion potential, (v)Flow-accelerated Corrosion, (vi)Water radiolysis, and (vii)Future subjects. The summary of his presentation is given in the following paragraphs.

Photo 4 Prof. Fumio Kojima

The latest experiences with corrosion in the cooling systems of nuclear power plants were reviewed in his lecture. High temperature cooling water causes corrosion of structural materials, which often leads to adverse effects in the plants, *e.g.*, increasing shutdown radiation, generating defects in materials of major components and fuel claddings, and increasing the volume of radwaste sources. Corrosion behaviors are much affected by water qualities and differ according to the values of water qualities and the materials themselves. In order to establish reliable operation, each plant requires its own unique optimal water chemistry control based on careful consideration of its system, materials and operational history.

Electrochemistry is one of key issues that determine corrosion related problems although not the only issue. Most phenomena for corrosion related problems, *e.g.*, flow-accelerated corrosion (FAC), intergranular stress corrosion cracking (IGSCC), primary water stress corrosion cracking (PWSCC) and thinning of fuel cladding materials, can be understood based on an electrochemical index, *e.g.*, electrochemical corrosion potential (ECP), conductivities and pH. The most important electrochemical index, ECP, can be measured at elevated temperature and applied to in situ sensors of corrosion conditions to detect anomalous conditions of structural materials at their very early stages.

In his presentation, theoretical models was also introduced based on electrochemistry to estimate wall thinning rate of carbon steel piping due to flow-accelerated corrosion and corrosive conditions which determines IGSCC crack initiation and growth rate . And as the future subjects of the theoretical models related to corrosion problems, he stressed that standardization of the codes should be established based on V&V evaluation procedures

For further detail of his presentation，see his paper [1] in this issue.

Photo 5 Dr. Shunsuke Uchida

3.4 Introduction of AR research activities for decommissioning of NPP

In the presentation of Prof. Hiroshi Shimoda (Kyoto University), he first introduced “what is AR(augmented reality) technology", and then proceeded to the application of AR for the decommissioning work of Fugen NPP now progressing in Japan. In his presentation, he especially introduced the feasibility study of AR application for the planning of dismantling and conveyance, and the two fundamental technology development: The methods of (i) circular marker based tracking and (ii)natural feature point based tracking in changing environment are considered to be suited for the real working condition of plant decommissioning.

A research paper by Yan Weida [2] in this issue is partly related with Prof. Shimoda’s presentation.

4 Record of afternoon session (I)

There were two presentations in the first part of the afternoon session. The both are the introduction of the automation researches that have been conducted at Danish Technical University (DTU).The summaries of the both presentations are given in this chapter.

Photo 6 Prof. Hiroshi Shimoda

4.1 Short introduction of automation researches at DTU

Prof. Ole Ravn (DTU) introduced the Danish Technical University and the research activities at his belonged Department of Electrical Engineering. The summary of his presentation is given in the following paragraphs.

The DTU initiated in 1829 and the first rector was H.C. Aersted (also name of Unit in electro-magnetics). At present, the total number of students of DTU is ca. 9300 including master and PhD students of ca.2150.According to Leiden Ranking 2013, the ranking of DTU is No.1 in Scandinavia and No.7 in Europe.

The major elements of the Department of Electrical Engineering, DTU are (i)acoustic technology, (ii)hearing system, (iii)biomedical engineering, (iv) electro-magnetic systems, (v)electronics, (vi)center for electric power and energy, (vii)automation and control, (viii)health and people care device (medico, welfare), (ix)Production, energy and supply system (wind turbine, intelligent grid/power), and (x)vehicles, robotics and service systems (robotics, aerospace, engine control, rail system). And according to Prof. Ravn, the core competence and domain of the department are (i)feedback and control, (ii)information processing and information, and (iii)system engineering and modularity.

He then introduced many interesting research activities in that direction as listed below only the titles or keywords:

(i)Parametric resonance detection,

(ii)Modular playware,

(iii)Diagnosis and fault tolerant control,
(iv)Functional modeling of complex systems,
(v)Supervision, navigation and software framework for mobile robot,
(vi)Autonomous robot systems for cleaning and natty bampo,
(vii) Robot for meat industry, and
(viii)Apple orchard safe and reliable project.

Photo 7 Prof. Ole Ravn

4.2 Functional modeling of control systems

Prof. Morten Lind (DTU) gave a short introduction to MFM and its foundation in theories of action, followed by explanation on how the action theory has contributed to formalization of control functions and how it also will contribute to extensions of MFM addressing with some of the challenges issues. The summary of his presentation is given in the following paragraphs.

Previous research on Multilevel Flow Modeling has developed concepts and tools for representation and reasoning about goals and functions of complex automated processes. MFM is presently applied for modeling nuclear power plants and processes within the oil and gas sector for risk assessment and supervisory control. One of the powerful features of MFM is the ability to represent relations between the physical process and the control systems on multiple levels of functional abstraction. These relations are of importance in analyzing the causes and consequences of disturbances in complex automated processes.

MFM concepts have foundations in logical and semantic theories of action. According to Prof. Lind, general control function types can be classified into three types: (i) Type I: Direct control (loops with setpoint control and disturbance rejection), (ii) Type II: Start-up shut down and transition between modes, and (iii) Type III: Optimizing control.

Prof. Lind's current MFM research deal with the following three challenges:
(i)Using MFM for reliability assessment of NPP control systems. Of particular interest is the study of type I and type II interactions.
(ii)Formalizing the representation of operation modes. The challenge is here both to represent the modes themselves and the control actions which provide transition between modes.
(iii)Development of principles for reasoning about control. "Traditional" techniques to reason about control is based on models of behavior. MFM offer a complementary framework for reasoning about means and ends of control.

Photo 8 Prof. Morten Lind

5 Record of afternoon session (II)

There were four presentations in the second part of the afternoon session. The subjects were diverged from Fukushima Accident analysis, to reliability analysis for PWR, to control system design for small modular reactor, and finally to the designing of digitalized I&C + HMIT for Japanese PWR The summaries of the four presentations are given in this chapter.

5.1 Post-facta analysis of Fukushima Daiichi accident by simple physical model

Dr. Fumiya Tanabe (SOCTEX) presented his own analysis of Fukushima accident by using simple physical models related with decay heat, water level in the reactor, heat and hydrogen generation by Zr-water reaction, *etc*., in order to conjecture the plausible reactor melt phenomena behavior to interpret the radiation monitor data surrounding the Fukushima site and many places far from the site.

According to him, analyses were performed of the first core melt behavior of the Unit 1, Unit 2 and Unit 3 reactors of Fukushima Daiichi Nuclear Power Station on 11-15 March 2011. The analyses were based on a measured data investigation and a simple physical model calculation. Estimated data are time variation of core water level behavior, core material temperature and hydrogen generation rate. The analyses revealed the characteristics of accident process of each reactor. Analyses were also performed also of the re-melt (melt again) behavior in another chaotic period of 19-31 March 2011.

The details of his analyses are available from his published papers [3-5] with his paper in this issue. [6].

Photo 9 Dr. Fumiya Tanabe

5.2 Reliability analyses of PWR safety systems by the GO-FLOW methodology

Prof. Takeshi Matsuoka (Utsunomiya University) made a comprehensive report on the results of reliability analyses for the safety systems of AP1000 by GO FLOW. The summary of his presentation is given in the following paragraphs.

AP1000 is one of several advanced pressurized water reactors (PWRs) which are now under construction in China. The AP1000 employs passive safety systems, which is defined as a safety system whose operation is only relied on passive components. A passive component does not require any external input or energy for its operation, and only relies on natural physical laws (gravity, natural circulation, conduction, *etc.*).

The reliability of AP1000 passive safety systems were evaluated by the GO-FLOW methodology, and compared with active safety systems of conventional PWR plants.

The AP1000 safety systems have two main systems, passive core cooling system (PXS) and passive containment cooling system (PCCS), while the conventional PWR safety systems consist of containment spray system (CSS) and emergency core cooling system (ECCS) which are active safety systems, i.e., they have active components as water injection pump, motor operated valve, *etc.*

The PXS consists of the following subsystems: Passive safety injection system (PSIS), four stages of an automatic depressurization system (ADS), passive residual heat removal system (PRHRS). There are two modes of operation in the PCSS system, five modes in the PXS, and four stages of ADS operations.

The operations of these safety systems have dynamical characteristics, and make phased mission problems. The GO-FLOW methodology can be well adopted to analyze dynamical system behavior and phased mission problems. Also important information could be obtained by the common mode failure and uncertainty analyses.
These two kinds of analyses were also performed in the comparisons of AP1000 and conventional PWR Safety systems.

Prof. Matsuoka summarized the GO FLOW analyses as follows;
(i) Quantitative dynamic reliability analysis of AP1000 passive safety systems was successfully conducted to confirm how to evaluate the dynamic reliability of the related safety systems by GO-FLOW methodology,
(ii) The passive safety system's concept was evaluated to be more reliable than the PWR's active

safety systems,
(iii)The passive safety components depend on the two types of failure modes: Type A (structural failure (hardware failure), physical degradation) and Type B (functional failure due to blocking of intended natural phenomena),
(iv)The reliability of AP1000 PXS was calculated to be higher from the blow-down phase to IRWST gravity injection phase,
(v)Then it decrease discontinuously in the recirculation phase because the redundancies of injection subsystems only reduced into the recirculation sump and also due to the increase of the failure probability of components with time,
(vi)ADS system is a key safety system of AP1000 for the successful actuation of subsystems of PXS and PCCS comparatively to PWR plant.

The details of GO FLOW analyses with regards to his presentation are available from the papers listed in Refs. [7-13].

Photo 10 Prof. Takeshi Matsuoka

5.3 The automatic control design and simulation of reactor control system in Small Modular Reactor

Mr. Longtao Liao (NPIC) introduced his research work on the designing of control system for a Chinese Small Modular Reactor(SMR) at NPIC. The summary of his presentation is given in the following paragraphs.

There are demand of multipurpose utilization of nuclear power other than electric generation around the world. There are six countries where 13 small modular reactors are now under developments with ca. 45 proposals of new SMR concepts.

In China, the development and application of Small Modular Reactor (SMR) aims at electricity generation, heat supply and seawater desalination. The major characteristics of SMR in China is a pressurized water reactor which has integrated head package to integrate the once-through steam generator (OTSG) into the reactor pressure vessel. This is to eliminate primary loop pipeline and it becomes an important for the co-ordinated control system designing of (i)reactor output power control and (ii)control process of SG feedwater flow.

He presented the automatic control design methods of both the reactor power and feed water control system based on the characteristic of the reactor and the OTSG. Concerning the test bed for designing the control systems for SMR, he used RELAP5 for plant model, MATLAB for control system model, and SQL database to integrate them for the study of control system simulation. The simulation results were also presented to illustrate the performance of the control scheme.

For further detail of his presentation see his paper [14] in this issue.

Photo 11 Mr. Longtao Liao

5.4 Development of Mitsubishi computerized human machine interface and digital I&C system for PWR plants

Mr. Koji Ito (MHI) made a brief introduction of the fully computerized Human Machine Interface (HMI) system and digitalized Instrumentation and Control

(I&C) System of Mitsubishi Heavy Industries, Ltd. The summary of his presentation is given in the following paragraphs.

The development of computerized main control board in Japan had started in 1989 with the participation of plant operators, and the project of the full digital main control room for PWR had been conducted in 1998-2001 in MHI with the participation of PWR utilities. Wherein three times V & V test had been conducted ranging from full scope simulator, to performance check and finally to operator review, in order to establish the technical design guideline for new PWR, Advanced PWR and for plant modernization. In Japan three full computerized main control boards for PWR were adopted in 2009, for a new PWR Tomari Unit 3 in Hokkaido Electric Power Company and the plant modernization at Ikata Units 1 and 2 in Shikoku Electric Power Company.

With regards to the computerized HMI system development for the nuclear power plants in USA, the V & V tests which had been conducted by the participation of plant operators in US plants, and the technical reports of the computerized HMI system had been submitted for US NRC in 2008 and 2009.Although the large display panel (LDP) , diverse HSI panel and superviser console are same as those in Japan, introduction of computer-based operation procedure and the shift technical advisory console were special ones developed for US computerized HMI.

Photo 12 Mr. Koji Ito

There were ten invited presentations at the ISSNP2013, and the subject of the presentations ranges from (i)material corrosion issue of NPP components, (ii)application of augmented reality technology for NPP decommission, (iii)functional modeling method for plant control system, (iv) intrinsic understanding of Fukushima Daiichi accident phenomena based on simple physical model, (v) system reliability evaluation method for PWR safety system, (vi) automatic control system design for small modular reactor, and (vii) validation of computerized human-machine interface and digital I&C for PWR plant.

6 Concluding remarks

The 17th International Workshop on Nuclear Safety & Simulation Technology (IWNSST17) was held in January 21, 2014 at Kyoto University, in Kyoto, Japan. This one-day workshop was motivated to exploit advanced safety researches for nuclear power plant (NPP), by a unique synergetic collaboration of basically two different disciplines: material science and systems sciences.

This article provides the overview of the IWNSST17 with giving condensed summaries of all invited presentations given by international experts. All the PPTs presented at the workshop are available on the New and Report of Symbio Community Forum's website.

http://symbio-newsreport.jpn.org/?type=home&action=main

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