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Consultants' Meeting (Preparatory Meeting) on the IAEA Coordinated Research Project (CRP)

"Benchmark Analyses of Sodium Natural Convection in the Upper Plenum of the MONJU Reactor Vessel"

Hosted by

Japan Atomic Energy Agency (JAEA)

Tsuruga, Japan

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Consultants' Meeting (Preparatory Meeting) on the IAEA Coordinated Research Project (CRP) "Benchmark Analyses of Sodium Natural Convection in the Upper Plenum of the MONJU Reactor Vessel"

Hosted by the Japan Atomic Energy Agency (JAEA)

Tsuruga (MONJU MC Square Hall), Japan, on 11 May 2007

Chairman: Akira Yamaguchi (Osaka University)
Scientific Secretary: Alexander Stanculescu (IAEA)

Meeting Report

1. Introduction

The one-day Consultants' Meeting is taking advantage from the presence in Tsuruga, Japan of many experts interested in participating in the IAEA Coordinated Research Project (CRP) "Benchmark Analyses of Sodium Natural Convection in the Upper Plenum of the MONJU Reactor Vessel". It has thus been convened on 11 May 2007 in conjunction with the 40th TWG-FR Meeting, hosted by JAEA, and scheduled from 14 – 18 May 2008 in Tsuruga and Kyoto. Its objective is to start the preparation of the CRP that will be kicked-off, most probably, in late 2008.

The Consultants' Meeting was attended by:

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The objective of the Consultants' Meeting was to start preparation of the above mentioned CRP. IAEA presented the rationale for the CRP and its implementation plan. JAEA presented the experimental and analytical work already performed in the area of interest to the CRP. The experts representing the Member States interested in participating in the CRP presented their experience and expectations from the CRP. Finally, a preliminary work plan was established, thus providing a head-start for the kick-off Research Coordination Meeting (RCM) that is tentatively scheduled for September 2008.

2. Background Situation Analysis

MONJU is a 714 MWth (280 MWe) loop-type prototype fast breeder reactor. It is fuelled with mixed uranium-plutonium oxide fuel and cooled by sodium (three loops). Initial criticality was achieved in April 1994, and connection to the grid in August 1995. In December 1995, while performing operation tests with the reactor at 40% power, a sodium leakage incident occurred in the secondary sodium circuit. The reactor has remained shut down since then. JAEA (the former PNC) have performed an extensive investigation of the causes of the leakage incident and a comprehensive safety review. Based on a thorough licensing procedure, the permit for plant modifications (including, among other, various countermeasures against potential sodium leakage) was issued in December 2002 by the Ministry of Economy, Trade and Industry (METI). JAEA has started preparatory work for modification, as soon as the local Fukui Prefecture governor has granted approval in February 2005. The main modification work is in progress since September 2005. First criticality is expected to be achieved in the first quarter of 2008, followed by the start-up program. The main objective is to resume MONU operation as soon as possible and achieve the initial goals of demonstrating the operational reliability of a fast reactor power plant, in particular with regard to sodium handling technology.

Thermal hydraulics of hot pool forms an important part of sodium cooled fast reactor vessel design. With reference to MONJU, due to the limited (as compared to the present) computational capabilities available at the time when the reactor was designed (roughly three decades ago), the reactor vessel design work had to rely heavily on a large number of mock-up experiments and numerical analyses based on pessimistic assumptions. Large margins had to be incorporated into the definition of the relevant parameters to account for rather important uncertainties. While this approach was appropriate at that time, since MONJU's main mission (as a prototype) was to confirm the technical feasibility of the sodium cooled fast reactor concept, current emphasis in fast reactor design work is shifting towards the demonstration of the economical competitiveness of power plants equipped with sodium cooled fast reactors. Therefore, current reactor component (in this case vessel) design must make use of state-of-the-art simulation codes that were developed on the basis of sophisticated physical models, and are made possible due to the vast improvement in computational technology. Moreover, current simulation methods have vastly increased the application fields of these codes to, e.g. the precise description of the transport of the delayed neutron precursors from failed fuel subassemblies, the confirmation of thermocouple locations, the identification of potential zones of thermal striping, gas entrainment possibilities, the analysis of the oscillatory nature of thermal stratification effects, etc.

However, the new numerical analysis methods require extensive validation efforts, preferably through collaborative international efforts. Recognizing that an IAEA initiated CRP is particularly well suited to provide the necessary international framework for such a validation effort, the specialists from the Member States of IAEA's Technical Working Group on Fast Reactors (TWG-FR) felt strongly the need for this CRP and recommended IAEA to implement it. The major activities to be carried out under this CRP, i.e. analytical/numerical benchmark exercises and comparisons with experimental data obtained at MONJU, will provide a wide basis for the validation/qualification of codes/methodologies being employed by the various Member States. Ultimately, the outcome of the CRP will be a contribution to more accurate simulation methods allowing reducing the margins and delivering cost effective design solutions, with enhanced safety features.

3. Overall Objective

The overall objective of the CRP is to improve the Member States' numerical simulation capabilities of complex thermal hydraulics of sodium cooled fast reactors. It is felt that a necessary condition towards achieving this objective is a wide international validation effort of the data and codes currently employed for the simulation of the various physical effects involved in this field. Towards realising this, it is expected that the experts from the interested Member States will contribute by participating and applying to the common benchmark exercises the different methodologies being used by them. In the first stage, the benchmark exercises will focus on the numerical simulation of the natural convection phenomenon in the upper plenum of the MONJU reactor vessel, for which coolant temperature and flow data were measured during the original MONJU start-up experiments, and will be made available by JAEA to the CRP participants.

4. Specific Research Objectives

For the first stage of the CRP, JAEA will submit to the participants the experimental data relevant to the envisaged benchmark exercise, specifically the sodium thermal stratification measurements performed in the MONJU reactor vessel upper plenum during a plant trip test conducted on 1 December 1995 with the reactor at 45% thermal power level (corresponding to 40% electrical power level). The trip test was meant to deliberately introduce abnormality in the condenser as triggering event. The subsequent scenario played out according to the following sequence: turbine trip (closure of the steam stop valve) → reactor trip (insertion of all the control rods) → sodium pump trip (only pony motors driving the pump and ensuring heat removal).

Process parameters data (coolant flow rates and temperatures at various locations) were measured during the above mentioned transient. JAEA will provide these data to the CRP participants along with the detailed description of the relevant geometry and other initial conditions, as needed by the participants to perform the respective simulations. The participants will apply their own methodologies (computer codes, modelling approaches, assumptions/simplifications, boundary conditions, etc) in their numerical simulations. Inter-comparisons between the various calculation results and between the calculations and the experimental data, including the results of the original analyses performed by JNC with the help of the multi-dimensional computer code AQUA (published in 1997), will be performed by the CRP participants. Based on the results of these inter-comparisons, subsequent investigations by the participants in the CRP will identify remaining open issues and further R&D needs to resolve them.

Based on the outcome of the first stage of the CRP, the possibility to extend the activities of the CRP to benchmark analyses of tests planned during the upcoming MONJU start-up experiments will be investigated. By the same token, given expressed views of the CRP participants in the consultants' meeting, the possibility to extend the scope of the CRP to benchmark exercises based on experimental data necessary for validating the coupled thermal hydraulics of the hot pool and the remaining systems will be discussed in upcoming RCMs among the CRP participants, the TWG-FR members and the TWG-FR's Scientific Secretary.

5. Relationship to IAEA's Sub-Programme Objectives

The CRP will be implemented as a programmatic activity of the IAEA Project 1.1.5.2 "Technology Advances in Fast Reactors and Accelerator Driven Systems" starting with the IAEA Program and Budget Cycle 2008 – 2009. The Project 1.1.5.2 has the objective, among others, to enable Member States to make informed decisions on the development of new or advanced fast reactor designs, and to increase cooperation between Member States in achieving advances in fast reactor technology development through international collaborative R&D. Given its objectives, as stated in Section 3 of this Meeting Report, the CRP clearly responds to the objectives of the IAEA Project 1.1.5.2.

6. Action Plan (Activities)

Member States with past and/or ongoing fast reactor programs are invited to participate in the CRP. The following institutes in Member States and international organizations have informally indicated their interest in participating:

China	China Institute of Atomic Energy
France	Commissariat à l'Energie Atomique CEA
India	Indira Gandhi Centre for Atomic Research
Japan	Japan Atomic Energy Agency
Rep. of Kazakhstan	Kazatomprom
Rep. of Korea	KAERI
Russian Federation	State Scientific Centre Institute of Physics and Power Engineering (IPPE) Obninsk
Switzerland	PSI
USA	ANL
OECD/NEA, Paris	

Following the establishment of an international team by putting in place research agreements and contracts, JAEA discloses the detailed data (see Sections 3 and 4 of this Meeting Report) to all the participants in the CRP. During this first preparatory stage of the CRP, IAEA, JAEA and the participants exchange information mainly by electronic communication means. To achieve its objectives, the CRP will comprise, during its actual implementation phase, the following coordinating activities:

I. First (kick-off) RCM to identify lead organisations among the CRP participants for the various topics/work packages, produce an agreed upon list of detailed tasks as well as work plans and deadlines, identify responsibilities for competing tasks, and to establish an outline and responsibilities for completion of the final CRP report (NE Series publications report).

II. Second and third RCMs to review progress of technical work and NE Series publications report status, and identify needed improvements and/or modifications to the tasks and/or work plans. In particular, at the 2nd and 3rd RCMs, the participants will discuss the possibility to extend the activities of the CRP to benchmark analyses of tests performed during the on-going MONJU start-up experiments, as well as to benchmark exercises aiming at validating simulation methods of the coupled thermal hydraulics of the hot pool and the remaining systems (see Section 4 of this Meeting Report).

III. Fourth RCM (if necessary) to review the status of the technical wok and perform an overall review of the CRP results, provide the final input to the NE Series publications report and finalize the draft of the NE Series publications report, identify open issues and actions to resolve them, and outline the road ahead as well as the Agency's role.

The estimated duration of the CRP is 3 – 4years.

The schedule of near-term activities and actions arrived at in this consultants' meeting is as follows:

Submission of a short write-up by the participants in this consultants' meeting giving the details of their perspective and requirements with respect to the CRP, which will be included in the formal Meeting Report	**End May 2007**
The TWG-FR Scientific Secretary to release the Meeting Report as IAEA working material, incorporating the comments from the participants	**Mid June 2007**
JAEA to provide the consolidated experimental and other relevant data for the benchmark exercises in the form of a technical report, taking into account the participants' requirements	**1 July 2008**
Kick-off RCM (tentative)	**September 2008**

The sequence of activities is detailed in the chart below:

Activities	Year (0)	Year (1)	Year (2)	Year (3)	Year (4)	Year (5)
	2007	2008	2009	2010	2011	2012
1. Setting up the CRP team	X	X				
2. Convene 1st (kick-off) RCM		X				
3. Convene 2nd RCM			X			
4. Convene 3rd RCM				X		
5. Convene 4th RCM (if necessary)					X	
6. Issue of the final CRP report (NE publications series report)						X

Attachment 1

Contributions to and expectations from the CRP – Argonne National Laboratory (USA)

Argonne NATIONAL LABORATORY

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May 24, 2007

Dr. Alexander Stanculescu
International Atomic Energy Agency
Nuclear Power Division
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Dear Alex:

Thank you for the opportunity to attend the recent expert's meeting in Tsuruga to consider an IAEAE Coordinated Research Project (CRP) on "Benchmark Analyses of Sodium Natural Convection in the Upper Plenum of the MONJU Reactor Vessel." Based on the excellent presentations delivered by our hosts from JAEA, I believe that participation by Argonne in the proposed CRP will significantly benefit the U.S. reactor development program. If future programmatic and budgetary conditions will allow, I can assure you that Argonne will participate fully in the benchmark analysis.

For us, the chief benefit of the CRP will be validation of multidimensional fluid dynamics capabilities for analysis of outlet plenum temperature distributions. As reactor designers seek new fuel handling features to reduce costs, upper internal structure configurations are becoming more compact, and higher fidelity analysis techniques are required to assess thermal stresses.

Argonne currently has 1) a reactor systems analysis code with an experimentally-based model for plenum stratification, 2) the COMMIX code (parent of the JAEA AQUA code), and 3) commercial fluid dynamics analysis codes. It is anticipated that all or some combination of these capabilities will be employed to perform the CRP analysis.

With our participation in the benchmark analysis, Argonne will share the knowledge gained in the process and the results of our analyses. We look forward to the opportunity to join with our international colleagues to participate in this CRP.

Best wishes,

Jim

James E. Cahalan
Manager, Engineering Analysis Department

A U.S. Department of Energy laboratory managed by The University of Chicago

Attachment 2

Contributions to and expectations from the CRP – KAERI (Republic of Korea)

KAERI Activity Plan for MONJU CRP

Thank you very much for your invitation to the IAEA CRP on "Benchmark Analyses of Sodium Natural Convection in the Upper Plenum of the MONJU Reactor Vessel". In KAERI we have developed a system analysis code as well as a CFD code for analysis of fluid flow and heat transfer in the upper plenum of liquid metal reactor. To perform a benchmark analysis of sodium natural convection in MONJU reactor vessel will be a valuable chance for KAERI to validate the code by comparing the prediction results directly with the measured plant data. It will also be a basis to refine the reactivity feedback model of SSC-K code because the reactivity model due to a expansion of CRDL is closely related to the temperature distribution in hot pool. The analysis of natural convection (thermal stratification) in the upper plenum of liquid metal reactor is also a challenging problem for the CFD code developers since the thermal stratification is not well modelled by the usual turbulence model like the k-epsilon model. Thus, KAERI will certainly participate in the benchmark analyses, and the analyses will be carried out in two ways;

(1) Analysis of the Natural Convection with the CFD code

- Carry out computation for the benchmark problem using the CFD code.
- Validation of the CFD code using the experimental data.
- Investigation of the choice of turbulence model on the accuracy of the solution.

(2) Analysis of the Natural Circulation with the System Analysis Code SSC-K

- Analyses of the natural convection experiments with the SSC-K two-dimensional pool model.
- Evaluation and validation of the SSC-K pool model with respect to the measured data and other analyses.
- Identification of other issues and R&D needs.

Attachment 3

Contributions to and expectations from the CRP – IPPE (Russian Federation)

The simulation of the thermal hydraulics phenomena in the different components of nuclear reactors is a very important part of the design process, requiring major efforts. Often, the accuracy and quality of computer code predictions has a direct impact on the reliability and safety features of the reactors under development. In Russia, during last two decades, the GRIF code was and is widely used for the analysis of the thermal and hydraulics parameters of the Russian fast reactors. Moreover, GRIF will continue to be used for the substantiation of design solutions for future fast reactors. The proposed benchmark offers a good opportunity for updating and improving the GRIF code on the basis of its validation against high-quality experimental data, and through collaborative international efforts.

IPPE will contribute with the following activities:

1. Development of input data sets for the Russian thermal hydraulics code “GRIF”, simulating the geometry and conditions of MONJU upper plenum
2. Debugging of the simulation cases, calculation of the thermal hydraulics parameters (distributions for steady state conditions), and comparisons with experimental data
3. Calculation of the parameters measured during the MONJU start-up experiments (transient conditions) using the GRIF code, and comparisons with the results obtained by the other participants.

Attachment 4

Contributions to and expectations from the CRP – JAEA (Japan)

Katsuhisa YAMAGUCHI
FBR Plant Technology Unit
Advanced Nuclear System Research & Development Directorate
Japan Atomic Energy Agency

Dr. Alexander Stanculescu
International Atomic Energy Agency

Dear Alex:

Thank you for all your efforts in launching the Coordinated Research Project on "Benchmark Analysis of Sodium Natural Convection in the Upper Plenum of MONJU Reactor Vessel." JAEA is grateful that MONJU is able to provide the verification data for this benchmark analysis, and highly appreciates the proactive attitudes of the Member States. JAEA confirms its recognition that MONJU is an internationally important property and has the honor to contribute to technology developments in the fields of system dynamics and thermal hydraulic analysis.

JAEA continues to prepare and arrange the information for numerical analysis, including solutions of the open problems raised in the discussions in the consultancy meeting, to complete the technical report required to submit to the TWG-FR Scientific Secretary by July 2008.

JAEA, Central Research Institute of Electric Power Industry(CRIEPI) and Osaka University will jointly conduct a series of numerical analyses using AQUA(JAEA), CERES(CRIEPI), and other available codes in order to derive findings on:

1) Adequate modeling approach of complicated hydraulic obstacles in regard to trade-offs among CPU time, accuracy, effect of asymmetric geometry on velocity and temperature distributions of liquid sodium,
2) Turbulence models- laminar approach, isotropic vs. anisotropic turbulence models, and estimation of turbulence kinetic energy at inlet boundary.

Attachment 5

Contributions to and expectations from the CRP – IGCAR (India)

IGCAR has got the following computer codes for the thermal hydraulics of nuclear reactor and FBR in particular:

- THYC-2 D and 3D (in-house codes)
- TRIO (issued by CEA France under bilateral agreement)
- PHOENICS (commercial code)
- STAR-CD (commercial code)
- FLUENT and CFX (commercial code available with collaborative institutions)

Through systematic planned experimental programme both at IGCAR and other collaborative institutions in the country, involving tests on scales down models with water and sodium, the above codes are thoroughly validated for the simulation of issues such as thermal striping, thermal stratification, gas entrainment in the hot sodium pool etc specific to FBR. The experience gained so far in predicting the results of test data and also engineering experiments conducted on FBTR (40 MWt loop type reactor) have given high confidence in using these codes. A team of about 20 engineers at IGCAR along with many researchers from other institutions of the country (~20 per year) are working in the thermal hydraulics analysis using the above codes.

India is very keen in participation of pool hydraulics analysis of MONJU reactor with special emphasis on stratification in sodium. We would like to carryout the analysis in stages starting from simplified 2D-asymmetric geometry to complicated 3D geometry. The velocity distributions in the sodium in the primary pipe especially at the exit locations and variance in the free level boundary conditions will be studied in detail through 3D simulations. Various turbulence models (k-Є, LDS, Chung model, etc) will be applied, with the objectives of recommending:

- Suitable turbulence model
- Efficient analysis methodology with 2D and 3 D geometries along with appropriate oundary conditions (optimized w.r.t computer time and accuracy)
- Pool hydraulics at full power after validating the results corresponding to 40% power level. For this MONJU should supply relevant data
- The outcome of the study will help to establish matured analysis methodology for the thermal hydraulics of hot and cold sodium pools of pool type FBRs.